LA-UR-99-6330

Approved for public release; distribution is unlimited.

Title: **TRAC-M VALIDATION TEST MATRIX**

Author(s): B. E. Boyack, M. Straka, and L. W. Ward

Submitted to:

http://lib-www.lanl.gov/la-pubs/00818499.pdf

TRAC-M VALIDATION TEST MATRIX

by

B. E. Boyack,[a] M. Straka,[b] and L. W. Ward[c]

[a]Nuclear Systems Design and Analysis Group
Technology and Safety Assessment Division
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

[b]Scientech Inc.
910 Clopper Road
Gaithersburg, Maryland 20878

[c]Information Systems Laboratories
11140 Rockville Pike
Suite 500
Rockville, Maryland 20852

May 2001

TRAC-M Validation Test Matrix

ABSTRACT

This document briefly describes the elements of the United States Nuclear Regulatory Commission's (NRC's) software quality assurance program leading to code qualification and identifies and proposes specific tests for qualifying the modernized TRAC code (TRAC-M) for a broad spectrum of pressurized- and boiling-water reactor accidents and transients such that the requirements of the NRC's software quality assurance program are satisfied.

Verification is the process of ensuring that the products and process of each major activity of the software life cycle meet the standards for the products and objectives of that major activity. Examples of verification activities include formal major life-cycle reviews and audits, formal peer reviews, and informal tests such as unit and integration testing. Verification efforts are not discussed in this report.

Validation is the process of demonstrating that the as-built software meets its requirements. Testing is the primary method of software validation. We have subdivided the TRAC-M validation test matrix into four elements. The first set of validation activities compares code-calculated results with data from tests other than those employing experimental data, designated Other Standard Tests. The second set of validation activities compares code-calculated results with data from Separate Effect Tests. The third and fourth sets of activities compare code-calculated results with data from Component Effect Tests and Integral Effect Tests, respectively. The four elements identified above constitute the TRAC-M Validation Test Matrix.

CONTENTS

Page

CONTENTS (cont)

APPENDICES

Page

FIGURES

TABLES

TABLES (cont)

Page

TABLES (cont)

Page

TABLES (cont)

Page

EXECUTIVE SUMMARY

INTRODUCTION

This document briefly describes the elements of the United States Nuclear Regulatory Commission's (NRC's) software quality assurance program leading to software (code) qualification and identifies a test matrix for qualifying the modernized Transient Reactor Analysis Code (TRAC-M) to the NRC's software quality assurance requirements. Code qualification is the outcome of several software life-cycle activities, specifically, (1) Requirements Definition, (2) Design, (3) Implementation, and (4) Validation Testing. The major objective of this document is to define the TRAC-M Validation Testing effort.

WORKING CONCEPTS

We first present several concepts that are important to the remainder of the summary.

Validation Testing: The process that allows the sponsor to determine whether a software product complies with its requirements. Validation Testing demonstrates and assures that the code and its models and methods satisfy the code's design objectives and are both applicable to and qualified for usage in specified targeted applications.

Verification: The process of demonstrating that the products and process of each major activity of the software life cycle satisfy the objectives and standards set forth for that major activity. Examples of Verification activities include formal major life-cycle reviews and audits, formal peer reviews, and informal tests such as unit and integration testing. Verification activities are not discussed in this report

TRAC-M Validation Test Matrix: The collection of separate effect tests (SETs), component effect tests (CETs) integral effect tests (IETs), and other standard tests (OSTs) selected to ensure that all important code features, models, and integrated calculation capabilities are tested. To ensure completeness, we have taken four-element structured approach to identifying the individual tests to be included in the TRAC-M Validation Test Matrix. First, we have identified the basic equation models, flow-field models and engineering correlations, equipment component models, and special-purpose models in TRAC-M that must be validated. Second, we have identified local, component, and system level processes and phenomena that must be modeled by TRAC-M. Third, we have identified the set of targeted applications associated with plant type and event scenarios that must be modeled. Fourth, we have identified candidate tests for incorporation in the TRAC-M validation test matrix.

TRAC-M CODE

The TRAC-M code comprises operational features that are the user's interface with the code, mathematical models for the phenomena, components and equipment that make up the physical system, and numerical solution methods for the mathematical models.

Each of these structural elements comprises many individual subelements. Operational features include the basic input and output functions that make the code useful to the users. The mathematical models include

- basic equation models for fluid flow, heat conduction, and power generation (for example),
- flow field and engineering correlation closure models for mass, momentum, and energy exchange,
- models for physical equipment components such as the pressurizer (for example), and
- special purpose models for phenomena and equipment, such as countercurrent flow limiting and critical flow.

Numerical solution methods are associated with each of the mathematical models.

The contents of these basic TRAC-M structural elements are further expanded by category, subcategory, and model as described in Section 3.

PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT) USAGE

A PIRT identifies and ranks the processes/phenomena occurring in a particular plant during a particular transient scenario, e.g., plant event, transient, or accident. Three contemporary pressurized-water-reactor (PWR) PIRTs and BWR PIRTs covering a variety of accidents and transients were reviewed and summarized to develop a consolidated PIRT for PWR and BWR applications.

CODE VALIDATION

We have subdivided the validation element into four elements: validation tests using standards other than those that employ experimental data from OSTs and validation tests comparing code-calculated results with SET, CET, and IET test data.

Validation Using OSTs. This sub-element of validation contributes to code qualification by comparing code-calculated results with standards that do not employ experimental data. It encompasses tests of specific code features or functions; comparisons to equilibrium, concept problems with known outcomes, or analytical problems with known solutions; and problems to test the properties of the numerical solution methods. The other standard tests recommended for validation of TRAC-M are presented in Section 6 of this report.

Validation Using SETs. SETs generally focus on a few processes or phenomena within a single component test fixture. SETs are experiments in which a very limited number of physical phenomena are of interest and detailed, high-quality data are obtained. The SETs data recommended for validation of TRAC–M are presented in Section 7 of this report.

Validation Using CETs. CETs investigate behavior in a plant component. Comparisons of code-calculated predictions to data from CETs provide the mechanism for an important aspect of the code qualification effort. Comparisons to CET data are necessary to assess the capability of thermal-hydraulic (T-H) code to predict component-level processes identified in PWR and BWR PIRTs. The CETs recommended for validation of TRAC-M are presented in Section 8 of this report.

Validation Using IETs. IETs generally focus on multiple, coupled processes and components in facilities that have numerous hardware components. IET data are most useful for assessing performance and qualifying the integrated T-H code for its targeted applications. The IET data recommended for validation of TRAC-P are presented in Section 9 of this report.

TRAC-M VALIDATION TEST MATRIX

Given the four-coverage-element approach, we developed the test matrix presented in Sections 6-9. Relative to previous TRAC validation matrices, the TRAC Validation Test Matrix presented in this document places a much greater emphasis on validating individual TRAC-M models and methods using SET data, particularly fundamental test data. There are TRAC-M models for which no direct SET data exist (i. e., data do not exist that can be used directly to validate these models because the effect of the processes/phenomena that they model cannot be isolated). The most important of these models are associated with the interfacial transport processes for mass, momentum, and energy. The direct consequence of this circumstance is that validation must proceed at present by indirect means.

For this release of the document, candidate validation tests have been identified and recommended for PWR and BWR large-break loss-of-coolant accident phenomena only at the local, component and system level. Tests have also been recommended for a variety of PWR and BWR plant types and accidents and transients.

ACKNOWLEDGMENTS

The authors express their appreciation to Dr. F. Odar of the United States Nuclear Regulatory Commission for his thorough review and suggestions for improving this document.

D. Hughes, a co-author of an earlier version of this document ("TRAC-P Validation Test Matrix," LA-UR-97-3900, September 1997), was instrumental in bringing this document to publication.

Finally, but no less important, are the contributions of the editorial staff who worked on this report throughout its preparation: L. Rothrock and A. Mascareñas.

ACRONYMS

1D	One dimensional
2D	Two dimensional
3D	Three dimensional
ADS	Automatic depressurization system
ATWS	Anticipated transient without scram
B&W	Babcock & Wilcox
BEM	Basic equation model
BETHSY	Boucle d'Etudes Thermohydrauliques Système
BWR	Boiling water reactor
CCFL	Countercurrent flow limiting
CCTF	Cylindrical Core Test Facility
CE	Combustion engineering
CET	Component effect test
CHF	Critical heat flux
CISE	Centro Informazioni Studi Esperienze
CL	Component level
CSAU	Code scaling, applicability, and uncertainty
CSNI	Committee on the Safety of Nuclear Installations
ECM	Equipment component model
EOS	Equation of state
FIST	Full Integral Simulation Test
FFEC	Flow-field models and engineering correlations
FLECHT-SEASET	Full Length Emergency Cooling Heat Transfer-Separate Effects And Systems Effects Test
GE	General Electric
GERDA	Geradrohr Dampferzeuger Anlage
IET	Integral effect test
INEL	Idaho National Engineering Laboratory
JAERI	Japan Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LB	Large break
LOBI	Loop for blowdown investigation
LOCA	Loss-of-coolant accident
LOFT	Loss-of-fluid test
LL	Local level
LOSP	Loss of offsite power
LSTF	Large Scale Test Facility
MIST	Multiple-Loop Integral System Test
MSLB	Main steam line break
NEA	Nuclear Energy Agency
NRC	United States Nuclear Regulatory Commission
NSM	Numerical solution methods

ACRONYMS (cont)

ODE	Ordinary differential equation
OECD	Organization for Economic Cooperation and Development
OST	Other standard test
OTIS	Once-through Integral Systems
OTSG	Once-through steam generator
PIRT	Phenomena identification and ranking table
PKL	Primarkreislaufe
PWR	Pressurized water reactor
ROSA	Rig of Safety Assessment
SB	Small break
SCTF	Slab Core Test Facility
SET	Separate effect test
SETS	Stability enhancing two-step method
SGTF	Steam generator test facility
SGTR	Steam generator tube rupture
SL	System level
SPES	Simulatore PWR per Esperienze di Sicurezza
SPM	Special-purpose model
SSTF	Steam Sector Test Facility
T-H	Thermal-hydraulic
THEF	Thermal Hydraulic Experimental Facility
THTF	Thermal Hydraulic Test Facility
TLTA	Two-Loop Test Apparatus
TPFL	Two-Phase Flow Loop
TPTF	Two-Phase Test Facility
TRAC	Transient reactor analysis code
TRAC-B	TRAC-boiling water reactor version
TRAC-M	TRAC-modernized version
TRAC-P	TRAC-pressurized water reactor version
UMCP	University of Maryland, College Park
UPTF	Upper-Plenum Test Facility
W	Westinghouse

1.0. INTRODUCTION

Thermal-hydraulic (T-H) systems codes, hereinafter called T-H codes, are specifically designed for a variety of targeted applications. Among these applications are (1) reactor safety analyses for both operating and planned reactors, (2) audits of licensee's calculations, (3) analyses of operating reactor events, (4) analyses of accident management strategies, (5) support for test planning and interpretation, (6) support for probabilistic risk assessments, (7) design analyses, and (8) nuclear plant training and instrument and control simulators. Given the significance of the applications for T-H codes, both envisioned and realized, it is important that they be qualified for their intended applications. Validation Testing demonstrates and ensures that the code and its models and methods satisfy the code's design objectives and are both applicable to and qualified for use in specified targeted applications.

1.1. Background

The United States Nuclear Regulatory Commission (NRC) has established an overall goal of maintaining core competencies in thermal hydraulics, reactor physics, and T-H codes to support regulatory decisions and the continuance of international exchanges. The NRC has elected to implement its T-H code development goals in a single code by executing the five-component development plan shown in Fig. 1-1. The Transient Reactor Analysis Code (TRAC)-Pressurized Water Reactor Version (-P), or TRAC-P, has been selected by the NRC as the base code for its T-H code development efforts. The current name for the single code under development is the modernized TRAC (TRAC-M) code.

1.2. Document Objectives

The objectives for this document are as follows:

- Briefly describe the elements of the NRC's software quality assurance program,[1-1] including validation efforts.

- Describe the concepts providing the foundation for development of the TRAC-M validation test matrix.

- Identify and propose specific validation tests for TRAC-M qualification that satisfy the requirements of the NRC's software quality assurance program. The set of tests thus identified constitutes the TRAC-M Validation Test Matrix.

1.3. TRAC-M Validation Test Matrix Concepts

TRAC-M is a state-of-the-art, best-estimate, transient, system analysis computer code for analyzing geometrically complex multidimensional T-H systems, primarily nuclear power plants. TRAC-M also can perform containment system analyses. However, this is a recently added capability; the containment features of the code are not treated in this release of the TRAC-M validation test matrix.

The TRAC-M computer code consists of two major functional elements. The first element consists of the individual, fundamental building blocks for the code. Examples of these building blocks are mathematical models of specific physical processes, such as heat conduction in a pipe wall or the friction between a moving fluid and the wall as fluid moves through a pipe. The former is a complete theoretical model, whereas the second requires experimental data to effect an engineering solution. The experimental insights are embodied in closure models, also called constitutive models. TRAC-M contains more than a hundred of these individual theoretical and closure models.

Taken one at a time, these building block models cannot simulate complex, multi-feature physical processes, e.g., the transient, systemwide, multiphase, thermal-hydraulic, and neutronic processes that arise in nuclear plants during accident and transient conditions. These models must be brought into a unified structure and must be integrated. Thus, the second element consists of the features that integrate the individual theoretical and closure models within the TRAC-M code such that it can be used for the broad applications to which it is targeted. Two primary integrating elements of the code are the basic two-phase equations describing mass, momentum, and energy transport and the numerical methods employed to obtain numerical solutions to these coupled transport equations and the building block models described above.

Within a nuclear power plant, as it undergoes either a transient or accident, processes are observed to occur at three phenomenological levels: the local level (LL), component level (CL), and system level (SL). Examples of local-level processes are interfacial heat and mass transfer, fluid shear at a fluid-wall interface, and fluid-to-surface heat transfer. Examples of component-level processes are coastdown of the reactor coolant pumps, liquid levels within a component, and multidimensional flows within a component. Component-level processes arise from a combination of local-level phenomena and processes. Examples of system level processes are oscillations, loop-to-loop asymmetries, and natural circulation. As with component-level processes, system-level processes arise from a combination of phenomena and processes at both the local and component level.

Clearly, if the TRAC-M code is to fulfill its design objectives, it must model the important phenomena and processes occurring at the local, component, and system levels. However, all phenomena and processes occurring within a nuclear power plant, whether at the local, component, or system level, do not have the same impact on the path and outcome of the accident or transient. Some phenomena and processes are more important than others in this regard. It is from this reality that the value of phenomena identification and ranking tables (PIRTs) derive. The essence of a PIRT is captured in its name: it first identifies all the processes and phenomena occurring in a specified nuclear power plant undergoing a specific accident or transient. It next ranks the identified processes and phenomena for importance relative to one or more primary evaluation criteria. The TRAC-M validation matrix uses all available pressurized-water-reactor (PWR) and boiling-water-reactor (BWR) PIRTs to construct a consolidated list of highly important processes and phenomena for which the adequacy of the TRAC-M code must be validated, including all LL, CL, and SL processes appearing in the consolidated PWR and BWR PIRT. PIRTs are the first driver in constructing the TRAC-M validation test matrix.

The code must also model a variety of plant types, e.g., Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse (W) PWRs, a variety of General Electric (GE)-designed BWRs, and the individual designs of each of these vendors. For example, there are lowered-loop and raised-loop B&W designs, System 80 and System 80+ designs by CE, and two-loop, three-loop, and four-loop W designs. Core designs may also vary between different units within the same category, e.g., W four-loop and GE BWR/4 designs. For each of the above vendor, plant type, and category features, the code must be able to predict the behavior of the plant accurately under both accident and transient conditions. Accidents to be simulated include a spectrum of loss-of-coolant accidents (LOCAs), steam-generator tube ruptures, and main steam-line breaks. Transients to be simulated include pressurization, depressurization, and reactivity increases. The requirement to simulate a variety of plant, accident, and transient types adequately are requirements on the system-level or integrated performance of the code. It is not sufficient that a particular local-level phenomenon or component processes be well simulated if the simulation of key system-level parameters is inadequate. Plant design and targeted applications are the second driver in constructing the TRAC-M validation test matrix.

The final requirements on the TRAC-M validation test matrix derive from the need to represent and simulate accurately the highly important local-, component-, and system-level phenomena and processes identified by the PIRTs and system-wide processes associated with the targeted plant designs and applications.

1.4. Document Structure

The report contains nine sections. We have endeavored to provide brief, yet complete, coverage of the topics in each section. Where additional coverage is deemed necessary to demonstrate completeness, we have provided the needed information in appendices.

Section 2 of this report provides an overview of code qualification, as implemented by the NRC's software quality assurance program. Section 3 provides an overview of the current release version (Version 3.0) of TRAC-M. Section 4 culminates with a consolidated PIRT for the phenomena expected to occur during PWR and BWR accidents and transients. Each phenomenon is cross-correlated to the appropriate TRAC-M model previously defined in Section 3. Section 5 identifies the plant, accident, and transient scenarios that constitute the current set of targeted applications for the TRAC-M code.

Sections 6–9 describe the tests selected for the TRAC-M validation test matrix. Section 6 identifies validation tests other than those employing experimental data; these are designated Other Standard Tests (OST). Section 7 identifies the separate effect test (SET) data selected for the TRAC-M Validation Test Matrix; Section 8 identifies the component effect test (CET) data; and Section 9 identifies the integral effect test (IET) data. The relationship between the PIRT driver, plant and application driver, and the TRAC-M validation matrix is illustrated in Fig. 1-2.

The appendices contain either conceptual or detailed supporting information for the TRAC-M validation test matrix.

REFERENCES

1-1. Division of Information Support Services, "Software Quality Assurance Program and Guidelines," NUREG/BR-0167 (1993).

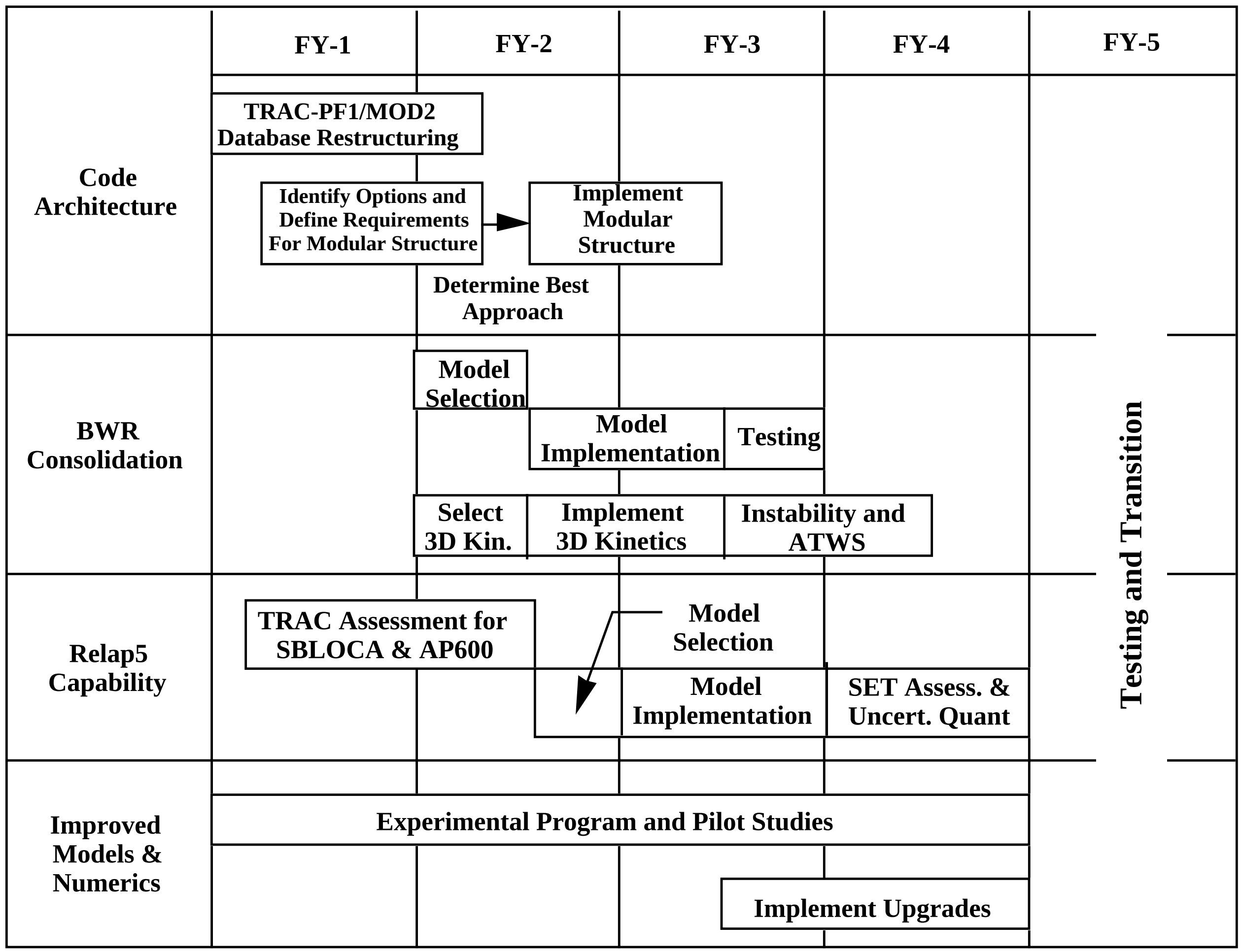


Fig. 1-1. Components of NRC's Thermal-Hydraulic Research Plan.[1-1]

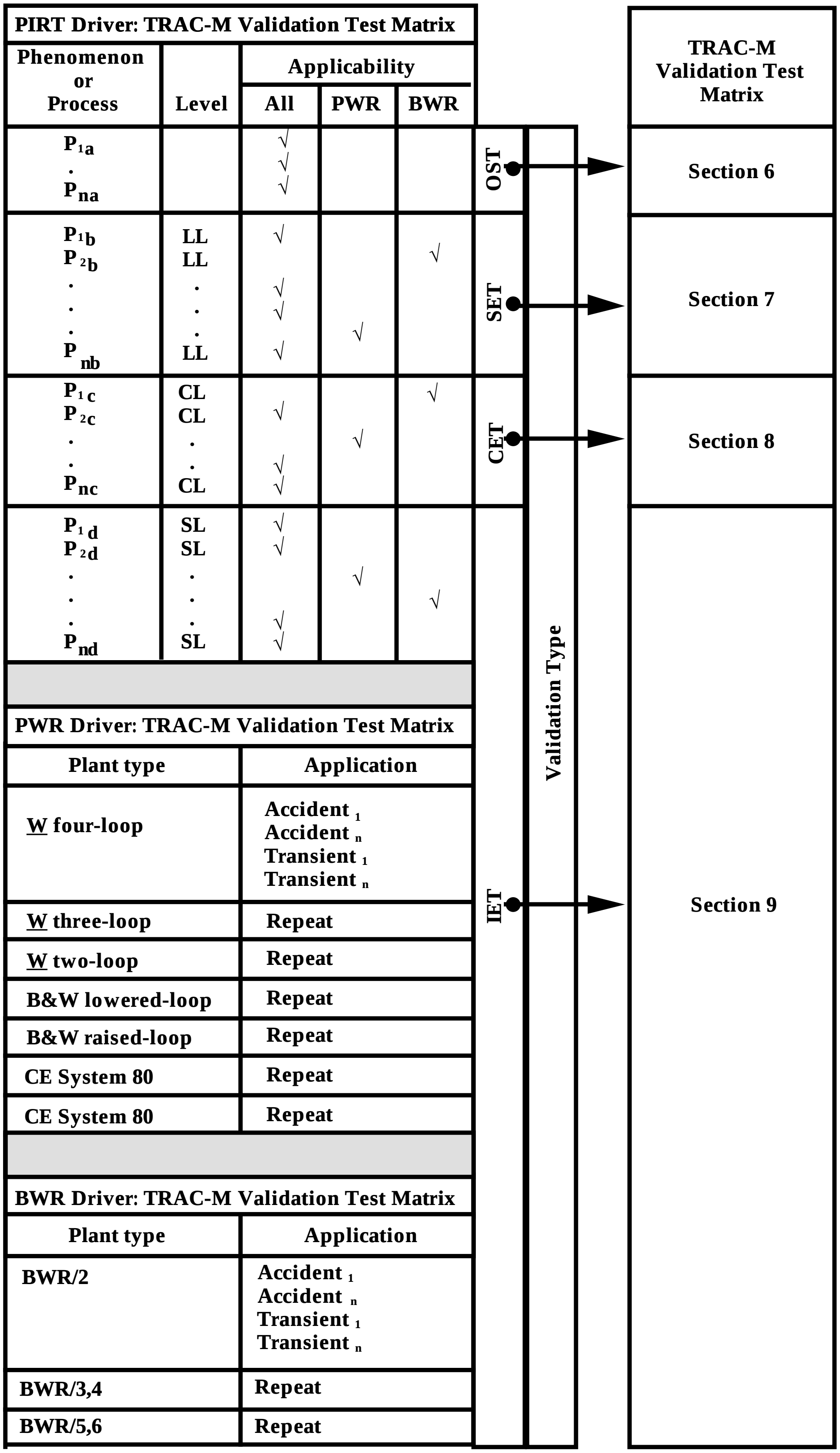


Fig. 1-2. Relationships of PIRT and plant and targeted applications to TRAC-M validation test matrix.

2.0. CODE QUALIFICATION OVERVIEW

Qualification is the process that allows the sponsor to determine whether a software product complies with its requirements. Completion of this process demonstrates and ensures that the code and its models and methods satisfy the code's design objectives and are both applicable and adequate for the specified targeted applications.

2.1. Code Qualification

Code qualification is the outcome of specific software life-cycle activities. The subset of software life-cycle activities culminating in code qualification is illustrated in Fig. 2-1. These activities are identical to those listed in Refs. 2-1 and 2-2. The life-cycle activities leading to code qualification are Requirements Definition, Design, Implementation, Verification, and Testing.

The life-cycle activities covered in Refs. 2-1 and 2-2 and shown in Fig. 2-1 assume creation and qualification of an entirely new code. Clearly, that is not the case for TRAC-M. Nevertheless, all of the life-cycle activities leading to code qualification will be described briefly here. The current status of TRAC-M within its software life-cycle is discussed in Section 3.6. The life-cycle activities are directed to the development of the following products: Requirements Definition, Design, Implementation and Testing.

- *Requirements Definition* is the set of activities that results in the specification, documentation, and review of the requirements that the software product must satisfy, including functionality, performance, design constraints, attributes, and external interfaces. The requirements form the basis for the software plans, products, and activities. Requirements should be necessary, complete, verifiable, consistent, unambiguous, modifiable, traceable, and technically feasible. Acceptance criteria that satisfy these requirements are defined during this life-cycle activity.

- *Design* is the set of activities that results in the development, documentation, and review of a software design that meets the defined requirements. Software design documentation specifies the overall structure of the software so that it can be translated into code.

- *Implementation* is the set of activities that produces the software. Implementation activities are conducted so that the software is developed in accordance with the design documentation and coding standards. It also includes informal unit and integration testing.

- *Testing* is the set of activities associated with formally testing, reviewing, analyzing, and documenting software performance.

Software quality assurance requires verification and validation of life-cycle products. The documentation that accompanies these software life-cycle activities is shown in Fig. 2-1 and is described further in Ref. 2-1.

- *Verification* is the process of ensuring that the products and process of each major activity of the software life cycle meet the standards for the products and the objectives of that major activity. Examples of verification activities include formal, major life-cycle reviews and audits, formal peer reviews, and informal tests such as unit and integration testing.[2-1]

- *Validation* is the process of demonstrating that the as-built software meets its requirements in accordance with selected acceptance criteria (success metrics). Testing is the primary method of software validation. The objectives of validation are to ensure that

  1. the as-built software correctly and adequately performs for all intended functions, e.g., targeted applications;

  2. the software does not perform any unintended function, either by itself or in combination with other functions that can degrade the entire system; and

  3. all nonfunctional requirements, e.g., performance, design constraints, attributes, and external interfaces, are met.

We have subdivided the validation effort into four elements: validation tests using OSTs, validation tests comparing code-calculated results with data from SETs, validation tests comparing code-calculated results with data from CETs, and validation tests comparing code-calculated results with data from IETs. This document provides a detailed description of the OSTs, SETs, CETs, and IETs that comprise the validation test matrix.

- *Validation Using OSTs.* This element of validation compares code-calculated results with standards that do not employ experimental data. It encompasses tests of specific code features or functions; comparisons to equilibrium, concept problems with known outcomes, or analytical problems with known solutions; and problems to test the properties of the numerical solution methods. An example of the first category, testing of code features, is a test to ensure that the input deck error checking is performing as designed. An example of the second category, equilibrium problems, is a test created by inducing a small imbalance in a U-tube manometer, followed by a return to equilibrium. An example of the third category, concept problems, is a test that checks whether the code returns a symmetrical result for a demonstrably symmetrical configuration. An example of the fourth category, analytical problems, is a comparison of code-calculated conduction results with the exact solution. An example of the fifth category, numerical method tests, is a problem that helps to characterize numerical diffusion.[2-3]

- *Validation Using SETs.* This element of validation compares code-calculated results with SET data. SETs are experiments in which a limited number of physical phenomena of interest occur and detailed, high-quality data are obtained under closely controlled conditions. SETs cover a spectrum of tests (Fig. 2-2), from the most fundamental to those investigating interactions between phenomena and components or equipment in a specific region of the physical system. Ideally, the fundamental, high-quality data should be used and the desired parameter measured directly. However, inherent to the basic two-fluid modeling approach used in TRAC-M is the requirement to provide closure models for wall-to-phase and interfacial heat, mass, and momentum exchange. This is a most challenging and difficult requirement because few complete and directly applicable sets of experimental data are available on which to base the mechanistic modeling of these exchange processes. Given this circumstance, only indirect validation at best is currently possible. The Organization for Economic Cooperation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) has produced the most comprehensive review of SETs facilities.[2-4] The primary use of data from SETs is to assess the adequacy of the closure relationships used in the code. These data also are used to address scaling issues. Because code predictions are compared with data, the definition of a precise set of performance measurement standards or success metrics is essential. Such a set of success metrics has recently been used in the qualification of the RELAP5 code for AP600 small-break (SB) LOCA analyses.[2-5] We subscribe to these success metrics (see Appendix A). The selected SETs become part of the validation test matrix. Additional perspectives regarding SETs are presented in Appendix B.

- *Validation Using CETs.* This element of validation compares code-calculated results with data from CETs, including transients measured in real plants. CETs investigate behavior in a plant component, frequently (but not always) at full scale (Fig. 2-2). Comparisons of code-calculated predictions to data from CETs provide the mechanism for an important aspect of the code qualification effort. Comparisons to CET data are necessary to assess the capability of T-H code to predict component-level processes identified in PWR PIRTs. In this manner, CET data are used to determine whether the behavior of the integrated code (e.g., field equations, closure relations, component models, numerics, and special models) are adequate at the CL. Component testing can occur in either SET or IET facilities.

- *Validation Using IETs.* This element of validation compares code-calculated results with data from IETs, including transients measured in real plants. IETs investigate behavior in a full nuclear power plant, usually in a reduced-scale facility (Fig. 2-2). Comparisons of code-calculated predictions to data from IETs provide the mechanism for three important validation efforts. First, comparisons to IET data are necessary to assess the capability of T-H codes to predict system-level processes identified in PWR PIRTs. In this manner, IET data are used to determine whether the behavior of the integrated code (e.g., field equations, closure relations, component models, numerics, and special models) are adequate. Second, IET data are selected to ensure that the code-targeted applications are represented (i.e., plant types and accident scenarios). Third, IET data are selected to address scaling issues. If possible, the selected IET facilities should cover a sufficiently broad spectrum of facility scales and transient types to support arguments of code applicability for plants. The OECD / NEA / CSNI has produced a comprehensive review of IETs facilities.

Deficiencies exist in the current TRAC-M code,[2-6] some of which are associated with the use of heuristic models in the code. Numerous others are associated with use of specific engineering correlations (closure models) beyond the range of applicability justified by their pedigree. Given this reality, code validation using IET data provides confidence that the resultant integrated code adequately predicts real plant performance. Once again, we subscribe to the success metrics (see Appendix A) that have recently been used in the qualification of the RELAP5 code for AP600 SB LOCA analyses.[2-5] The selected IETs become part of the validation test matrix. Additional perspectives regarding IETs are presented in Appendix B.

Taken together and properly implemented, these elements (Requirements, Definition, Design, Implementation, and Testing) provide the basis for qualifying a code for its targeted applications.

2.2. Validation Test Matrix

Information from several sources is needed to create a comprehensive TRAC-M validation test matrix, as shown in Fig. 2-3. These sources include information about the TRAC-M models and about processes and phenomena occurring during plant events and accidents in PWR and BWR plants. The various test problems and experimental data needed to complete the validation test matrix are discussed in Sections 6–9.

A formal release version of the code, i.e., release of a fully qualified code and associated documentation, always should be preceded by full-scope testing of the code against the validation test matrix. Although there is no set interval between two formal release versions of a code, the time and effort expended to qualify the code are such that 2 years between formal releases is probably the minimum, with the norm approaching 3 years.

2.2.1. Data Characterization

An essential element of data selection is data characterization. The important characterizing factors are as follows:

- experiment characteristics,
- applicability of data,
- data availability,
- quality of data, and
- range and variety of data.

The first factor, experiment characteristics, focuses on the experimental scale, instrumentation, and availability of information to develop a database from which a facility input deck can be prepared. The second factor, applicability of data, focuses on phenomena and the associated code models, specifically those identified in the summary PIRT (Section 4, Table 4-5). This factor addresses whether the data can be used directly to validate a particular model or whether they can be used only in an indirect manner to infer the characteristic behavior of the model. This factor also addresses whether the data are fundamental or derived from single or several components test facilities. The third factor, data availability, addresses whether the data can be acquired. The fourth factor, quality of data, is evident; high-quality data are required if the validation part of code qualification is to reflect code capabilities and adequacy accurately. An important measure of quality is the extent to which the data have been accepted and used for other code validation efforts. The fifth factor, range and variety of data, addresses the pragmatic issue of the cost of preparing facility input decks. Given two SET facilities, which are equal in all aspects except that a broader range of conditions is covered in one, we would select the facility with the broader range and variety of data because overall program costs are reduced.

2.2.2. Existing TRAC-M and RELAP5 Models

For some specific model validation efforts, there are several candidate facilities and data sets from which to choose. For example, numerous facilities have simulated film boiling; therefore, choices must be made. For this initial release of the validation test matrix, our selections are made using the following selection criteria:

- Facilities for which up-to-date TRAC-M input decks exist are given priority.

- Facilities for which TRAC-M input decks for earlier code versions exist are assigned the next highest priority; the input decks must be updated to run on the latest code version.

- Facilities for which RELAP input decks and a sufficient document database exist to permit creation of a TRAC-M input deck are assigned the next highest priority.

2.2.3. Data Sources

Various sources of information have been used to identify potential SET validation tests, including the following.

- The OECD / CSNI compilation of 185 SET facilities.[2-4]

- Reports on validation of TRAC-M and other computer codes (Refs. 2-7 through 2-11).

- Electronic bibliographies of publications associated with the TRAC-M, RELAP5, and RETRAN computer codes.

- Citations identified as a result of performing computer-based searches of the scientific literature.

2.4. Standard Test Matrix

Because there is an extended interval between formal release versions, numerous interim versions of the code are created during the interval. Interim versions are created to incorporate on-going code modification or development efforts, user enhancements, and error corrections. Because numerous interim versions are anticipated, it is desirable to define a smaller matrix that tests many, but not all, code features, algorithms, and equations. The test matrix so defined is the Standard Test Matrix. It is a subset of the TRAC-M validation test matrix optimized in some manner to fulfill the contradictory requirements of maximizing coverage of code features, algorithms, and equations while minimizing the resource requirements, e.g., the number of problems to be calculated.

The Standard Test Matrix will not fulfill all testing needs for every interim version, e.g., when an enhanced or revised model is untested by the problems in the Standard Test Matrix. Thus, for each interim version, it will be necessary to review the assessment needs and define, if needed, additional specific tests for the modified code.

2.5. Completeness Issues

An important goal to be attained in developing the TRAC-M validation test matrix is that of complete coverage. Ideally, there should be complete coverage of all code features, algorithms, and equations while minimizing duplication.

One ideal of completeness is that the TRAC-M validation test matrix contains problems that represent all of the important plants, facilities, systems, components, processes, and phenomena that arise from the targeted applications for the code. This aspect of coverage is considered in Section 5.

A second ideal of completeness is that the TRAC-M validation test matrix exercises each elemental part of the code, the input, output, subroutines, and, indeed, every line of code. Software now exists to create this database.* With existing coverage software, it is possible to run individual problems within either the TRAC-M validation test matrix or the standard test matrix and determine which specific lines of code are activated by the problem. In addition, it is possible to combine the individual results to determine the lines of coding activated by any subset of the validation matrices or the totality of the validation matrices. This information can be obtained only by exercising (running) the code for each of the specific tests within the validation test matrix.

REFERENCES

2-1. *Software Quality Assurance Program and Guidelines,* US Nuclear Regulatory Commission Document NUREG/BR-0167 (February 1993).

2-2. F. Odar, “Software Quality Assurance Procedures for NRC Thermal Hydraulic Codes,” internal US Nuclear Regulatory Commission document RPSB-99-1 (January 1999).

2-3. P. F. Peterson, “A Method for Predicting and Minimizing Numerical Diffusion,” *Numerical Heat Transfer,* Part B, V. 21, pp. 343–366 (1992).

---

* One example is Pure Atria Corporation’s PureCoverage™ software, which provides a precise and accurate way to gather code coverage data. This and like software provide a means to identify what parts of the program were and were not tested.

2-4. “Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, Volume I, Phenomena Characterization and Selection of Facilities and Tests; Volume II, Facility and Experiment Characteristics, Committee on the Safety of Nuclear Installations,” OECD Nuclear Energy Agency report NEA/CSNI/R(93)14/Part 1, Part 2/Rev. (September 1993).

2-5. C. D. Fletcher, P. D. Bayless, C. B. Davis, M. G. Ortiz, T. K. Larson, S. M. Sloan, R. A. Shaw, R. R. Schultz, C. E. Slater, G. W. Johnsen, L. S. Ghan, and D. E. Bessette, “Adequacy Evaluation of RELAP5/MOD3, Version 3.2.1.2, for Simulating AP600 Small-Break Loss-of-Coolant Accidents (Final Draft),” Idaho National Engineering Laboratory document INEL-96/0400 (December 1996).

2-6. B. E. Boyack, “TRAC-PF1/MOD2 Adequacy Assessment—Closure and Special Models,” Los Alamos National Laboratory draft document LA-UR-97-232 (February 21, 1997).

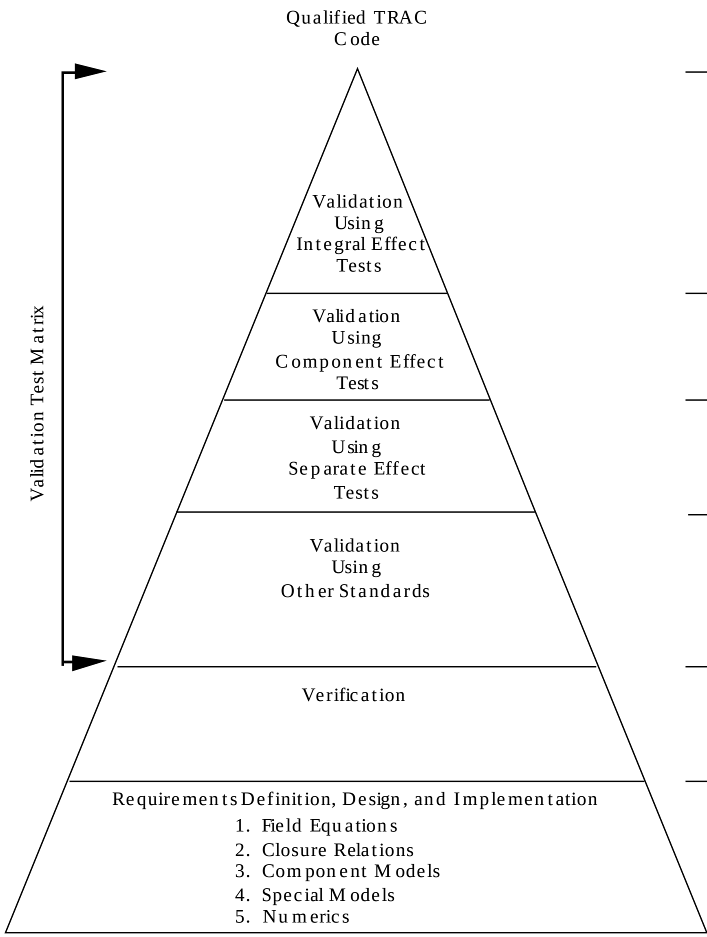
2-7. R. G. Steinke, “A Description of the Test Problems in the TRAC-P Standard Test Matrix,” Los Alamos National Laboratory report LA-UR-1475 (1996).

2-8. B. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore and J. L. Steiner, “TRAC-M: Fortran 77, Version 5.5, Developmental Assessment Manual, Volume I: Nonproprietary Assessment Sections,” Los Alamos National Laboratory document LA-UR-99-6480 (December 1999).

2-9. T. D. Knight, “AP600 Large-Break Loss-of-Coolant-Accident Developmental Assessment Plan for TRAC-PF1/MOD2,” Los Alamos National Laboratory report LA-UR-96-2700 (1996).

2-10. M. P. Paulsen et al., “RETRAN-3D—A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems Volume 4: Verification and Validation,” Electric Power Research Institute report EPRI NP-7450 (1996).

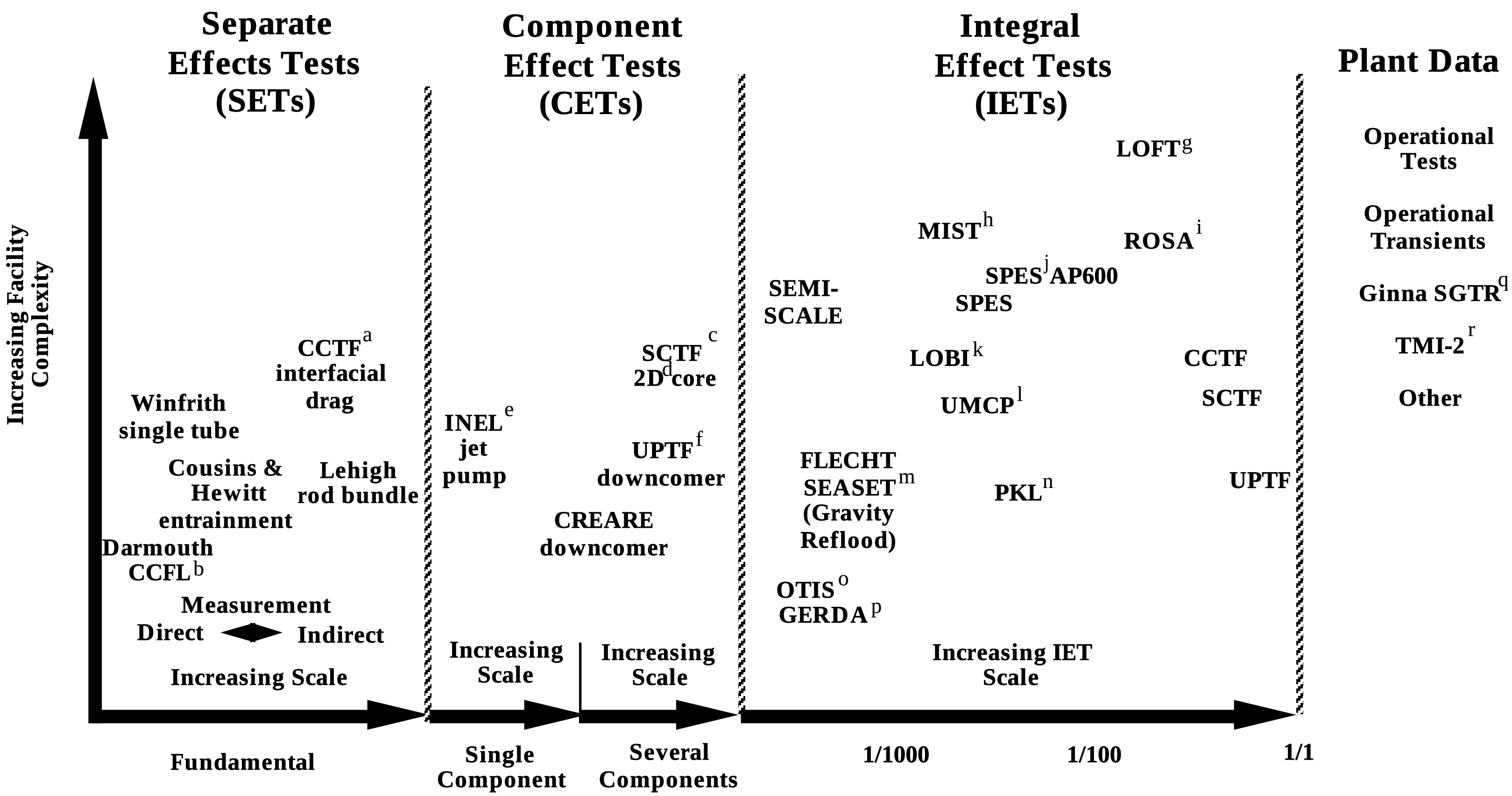
2-11. K. Carlson, R. Riemke, S. Rouhani, R. Shumway, W. Weaver, “RELAP5/MOD3 Code Manual Volume III: Developmental Assessment Problems (DRAFT),” C. Allison, C. Miller, N. Wade, Eds., EG&G Idaho, Inc. document EGG-2596, NUREG/CR-5535 (June 1990).



Description	Documentation
Integrated Code Behavior • Field Equations • Closure Relations • Component Models • Special Models • Numerics Scenario Applicability Scaling to Plant Conditions	Validation Test Plan Validation Test Procedures Validation Test Report Acceptance Test Plan Acceptance Test Procedures Acceptance Report User Documentation
Testing is variously performed using SET, CET, or IET data, depending on availability and applicability	
Focus on Closure Relationships (e.g., Correct Equations Used) Few Phenomena/Processes Simple Geometry Controlled Boundary Conditions	
Specified Design Features/Functions Equilibrium Problems Concept Problems Analytical Problems Numerical Method Tests	
Reviews Audits Peer Inspections Informal Tests	Verification and Validation Plan Review and Audit Packages Peer Inspection Packages Informal Test Plans, Procedures, Results
Analyze Requirements Develop Preliminary Design Develop Detailed Design Develop Code Units Inspections Reviews	Requirements Documents Verification and Validation Plan Software Project Plan Software Design Documentation Validation Test Plan and Procedures Acceptance Test Plan and Procedures

* For additional information see NUREG/BR-0167, "Software Quality Assurance Program and Guidelines," US NRC (February 1993)

Fig. 2-1. Code qualification overview.*



Note: Figure is illustrative and is not intended to include all SET, CET, and IET facilities in the test matrix.

---

[a] Cylindrical Core Test Facility.
[b] Counter current flow limitation.
[c] Slab Core Test Facility.
[d] Two dimensional.
[e] Idaho National Engineering Laboratory.
[f] Upper-Plenum Test Facility.
[g] Loss of Fluid Test.
[h] Multiloop Integral Test Facility.
[i] Rig of Safety Assessment.
[j] Simulatore PWR per Esperienze di Sicurezza.
[k] Loop for Blowdown Investigation.
[l] University of Maryland, College Park.
[m] Full Length Emergency Cooling Heat Transfer-Separate Effects And Systems Effects Test.
[n] Primarkreislaufe.
[o] Once-Through Integral Systems.
[p] Geradrohr Dampferzeuger Anlage.
[q] Steam-generator tube rupture.
[r] Three Mile Island, Unit 2.

Fig. 2-2. Spectrum of SET, CET and IET facilities.

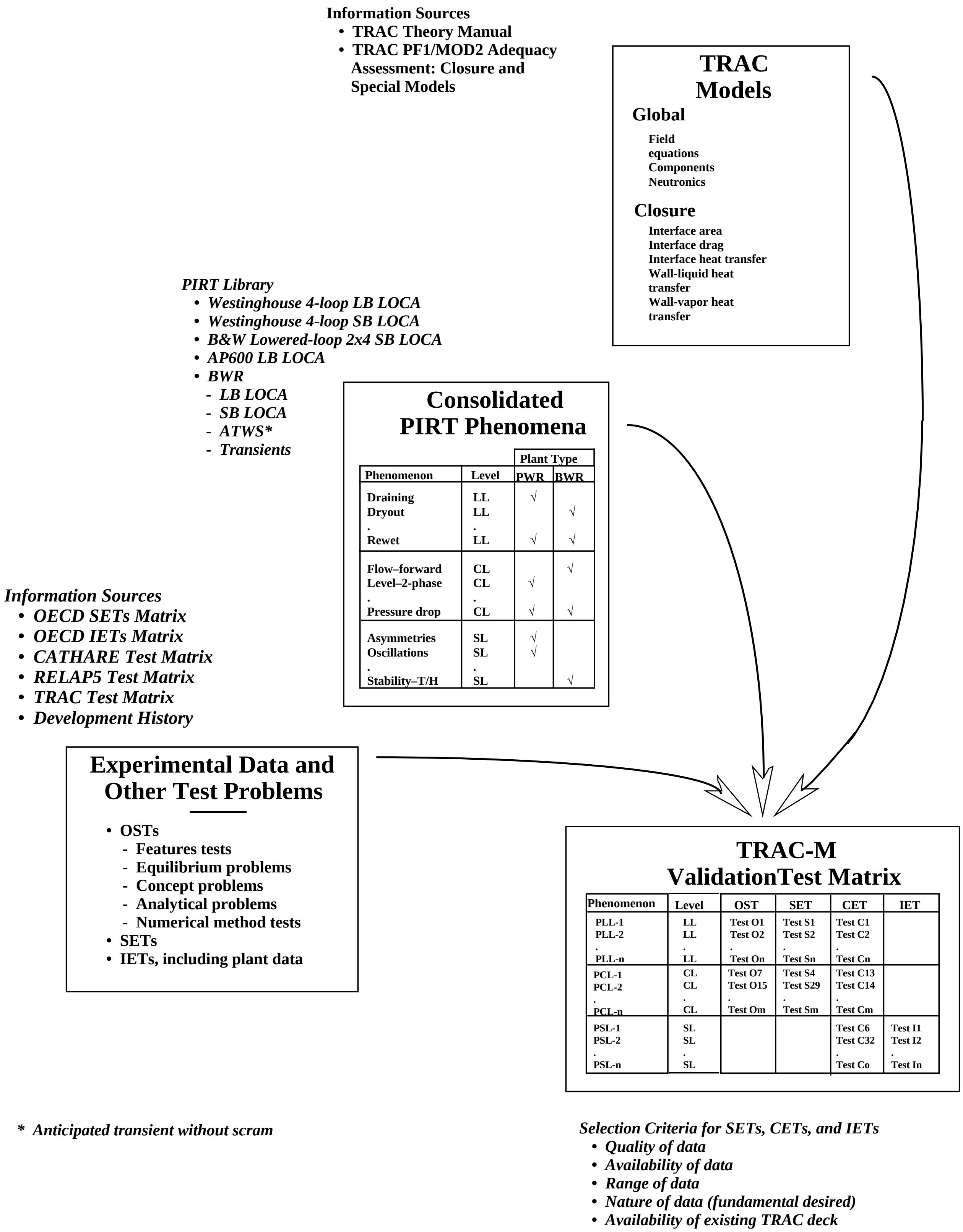


Fig. 2-3. Information sources supporting creation of TRAC-M validation test matrix.

3.0. TRAC OVERVIEW

The NRC is consolidating the capabilities of four of its T-H neutronics codes, i.e., TRAC P,[3-1]), TRAC-B,[3-2] RELAP-5,[3-3] and RAMONA,[3-4] into a single state-of-the art analysis code, TRAC-M. TRAC-M is a state-of-the-art, best-estimate, transient, system analysis computer code for analyzing geometrically complex multidimensional T-H systems, primarily nuclear reactor power plants. TRAC-M will be used by government and industry for design and safety analysis; phenomenological studies; operational transient analysis; evaluation of emergency operating procedures, simulator support and operator training; and assessment of data involving basic experiments, separate-effects tests, and plant operations. TRAC-M will calculate fluid flow involving gas, liquid, and mixture states in one-dimensional (1D) and three-dimensional (3D) rectilinear and cylindrical coordinates.

The TRAC-M computer code can be viewed as being based on two major theoretical elements. The first element is made up of the mathematical models that describe the physical processes / phenomena needed for the applications areas for which the code is designed. The second element is the numerical solution methods applied to the mathematical models. All aspects of both parts of TRAC-M must be tested during the verification, validation, and qualification procedures.

The mathematical models are further assigned to one of four categories, as shown in the following list:

1. basic-equations models (BEMs),
2. flow-field models and engineering correlations (FFECs),
3. equipment-component models (ECMs), and
4. special-purpose models (SPMs).

The details of the contents of the four mathematical model categories and the numerical solution methods (NSMs) are described further in the following paragraphs. The acronyms are defined to facilitate the information entered in various summary tables presented throughout the reminder of this document.

3.1. Basic Equation Models

The BEM category in TRAC-M includes the following subcategories:

- fluid mass,
- fluid momentum,
- fluid energy,
- noncondensable gas mass,
- dissolved solute in the liquid,
- 3D vessel,
- heat conduction,
- power generation in fuel,
- radiative energy exchange in the core,
- equation of state for fluids, and
- fluid thermophysical and transport properties.

Several of the subcategories are subdivided further into models. This decomposition of the BEM category into subcategories and models is presented in Table 3-1. This construct (category, subcategory, and model) is emphasized here because this format is utilized in Section 4 to cross-correlate the PWR and BWR PIRT phenomena and processes to TRAC-M models.

The fluid flow equations include mass, momentum, and energy equations for the vapor and liquid phases of the water plus mass conservation equations for noncondensable gases and dissolved solids. These model equations are applied in the 1D formulation to most of the physical system and in the 3D formulation for the reactor pressure vessel. A TRAC-M Fill component is used to apply a specified fluid velocity or flow at a boundary link, and a TRAC-M Break component is used to specify the pressure at a boundary.

The heat conduction model includes both 1D and 2D formulations for both rectangular and cylindrical solid structures. The 2D form generally is applied only to the modeling of reflood heat transfer in the fuel rods in the core. The conduction model can handle all three of the consistent boundary conditions for the parabolic heat conduction equation. A lumped-capacitance form of the conduction equation is also available.

The power generation in the core is modeled in three ways: the power can be (1) specified by the user, (2) modeled as point-kinetics decay heat, or (3) modeled by 3D neutron kinetics. Reactivity feedback is accounted for by changes in fuel and coolant temperature and coolant density. The power deposition in the fuel rods can be specified by the user as a function of position in the rod.

The 2D radiative energy exchange model is designed to handle radiative energy exchange between the heat structures assigned to hydro cells in a TRAC-M model of a physical system. The model includes accounting for the effects of a two-phase fluid mixture between the radiating surfaces.

The equation of state for water in TRAC-M uses the pressure and temperature as independent variables and returns all other fluid thermodynamic state properties plus various derivatives of these properties needed for the numerical solution methods. Properties for both the liquid and vapor phases are determined by polynomial fits to water property tabulations. All necessary thermophysical and transport properties for water are also available. The equation of state for the gases that can be included in the fluid flow model is based on the perfect gas model. The thermophysical properties of the gases are determined by derivatives of the equation of state, and transport properties are given by polynomial fits to data.

The material properties for the solid materials needed by the conduction equations are also available.

3.2. Flow Field Models and Engineering Correlations (Closure)

The basic fluid flow equations need various models to account for mass, momentum, and energy exchange between the flow-channel walls; between each phase in the flow field; and between the liquid and vapor phases. The models for these processes generally comprise correlations for heat, mass, and momentum exchange taken from the literature. These correlations account for the majority of the empirical correlations in the TRAC-M code.

The FFEC category in TRAC-M includes the following subcategories:

- regime maps
- fluid mass equation closure (mass exchange), including
  - subcooled boiling,
  - interfacial mass exchange, and
  - solute mass exchange;
- fluid momentum equation closure (momentum exchange), including
  - wall-to-phase momentum exchange,
  - interfacial momentum exchange, and
  - local pressure losses;
- fluid energy equation closure (energy exchange), including
  - wall-to-phase energy exchange and
  - interfacial energy exchange.

Although it is not clear that regime maps should be classified as closure models, they are so closely associated with the closure models that we have elected to include them with these models.

Several of the subcategories are subdivided further into models. This decomposition of the FFEC category into subcategories and models is presented in Table 3-1. This construct (category, subcategory, and model) is emphasized here because this format is utilized in Section 4 to cross-correlate the PIRT phenomena and processes to TRAC-M models.

In numerous cases, additional sublevels for the FFEC models are listed in Table 3-1. For completeness, these lower-level models are tabulated in Tables C-1 through C-6 in Appendix C. The information in Table 3-1 and Appendix C is extracted from Ref. 3-2. Verification and validation of TRAC-M ultimately will focus on the individual correlations given in Appendix C.

3.3. Equipment Component Models

Models for equipment components are usually developed and used when

- the equipment, and the phenomena that occur in the equipment, are so complex or too-little understood that a reliable mathematical description of the equipment and processes at a fundamental level is not possible; and

- the computational costs of using a more fundamental description of the equipment and processes would be too high for use in a systems-analysis computer code.

Equipment component models are usually based on an input-output type of model, and the details of the phenomena are not directly accounted for. The phenomena that occur in some equipment components require specialized modeling that cannot be easily obtained directly from the basic-equation models in TRAC-M.

The ECM in the TRAC-M code contains the following equipment components subcategories:

- centrifugal pumps (Pump component),
- jet pumps (Jetp component)
- steam-water separator (Sepd component),
- Plenum component,
- Valve component,
- turbine (Turb component), and
- pressurizer (Prizer component).

The ECM subcategories are not further subdivided into models; however, the decomposition of the ECM category into subcategories is repeated in Table 3-1 for completeness. This construct (category and subcategory) is emphasized here because this format is utilized in Section 4 to cross-correlate the PIRT processes/phenomena to TRAC-M models.

3.4. Special-Purpose Models

The SPM category in TRAC-M includes the following subcategories:

- countercurrent flow limitation model;
- critical flow model for fluid boundary conditions;
- trip and control system elements;
- reflood heat-transfer models, including
  - flow regime modeling,
  - wall-to-phase fluid drag,
  - interfacial fluid drag,
  - wall-to-phase fluid heat transfer,
  - interfacial fluid heat transfer, and
  - conduction heat transfer;
- two-phase mixture level tracking model;
- offtake model for Tee component; and
- fuel-cladding gap conductance.

With the exception of the reflood model, the SPM subcategories are not subdivided further into models. However, decomposition of the category into subcategories is repeated in Table 3-1 for completeness. The reflood heat-transfer model is subdivided further into models. This further decomposition of the reflood heat transfer subcategory into models is presented in Table 3-1.

In numerous cases, additional sublevels for the SPM are listed in Table 3-1. For completeness, these lower-level models are tabulated in Tables C-7 through C-9 in Appendix C. The information in Table 3-1 and Appendix C is extracted from Ref. 3-2.

3.5. Numerical Solution Methods

All of the mathematical models in the TRAC-M code must be integrated into the overall solution methods used to advance the model equations over a timestep. Generally, finite-difference approximations to the continuous equations are used to implement the solution methods. The resulting systems of algebraic equations are then solved to advance the time.

The NSM category in TRAC-M includes the following subcategories:

- fluid field equations, including
  - 1D stability enhancing two-step (SETS) method and
  - 3D SETS method; and
- conduction in solid materials, including
  - 1D rectangular and cylindrical,
  - 2D rectangular and cylindrical,
  - lumped capacitance method; and
- conduction boundary conditions;
- power generation in the fuel rods;
- trip and control system elements;
- fluid equation of state;
- fluid boundary conditions;
- equipment component models;
- special-purpose models;
- steady-state solution methods; and
- timestep size and control methods.

The steady-state solution methods have been developed to accelerate the solution of the transient equations to the steady-state condition. The timestep size and control methods are used to ensure the accuracy and stability of the solution method for the fluid flow equations.

The NSM subcategories are not subdivided further into models; however, the decomposition of the NSM category into subcategories is repeated in Table 3-1 for completeness. This construct (category and subcategory) is emphasized here because this format is utilized in Section 4 to cross-correlate the PIRT processes / phenomena to TRAC-M models.

3.6. Current Qualification Status

The TRAC-M code and it predecessors have been under development for approximately 25 years. Much of the rigorous structure and documentation envisioned in the NRC's software quality assurance program and guidelines, as summarized in Section 2.0, have not been realized. This is not to say that TRAC-M is found to be inadequate for its targeted applications. It is to state that its adequacy cannot be demonstrated to be in compliance with the NRC's software quality assurance program and guidelines. In the remaining paragraphs of this section, the current code qualification status of TRAC-M is reviewed briefly relative to each of the life-cycle activities leading to code qualification described in Section 2.1.

Requirements Definition, Design, and Implementation. Clearly, field equations, closure relations, component models, special models, and numerics have all been specified, selected, and incorporated into the present TRAC-M code. Some, but not all, of the documentation called for in the NRC's software quality assurance program and guidelines exist. However, requirements and specification documents, design reports, and independent review audits do not. A suite of TRAC-P documentation exists,[3-6–3-10] but a key document has remained in draft form for several years.[3-6] The primary code documentation is currently being updated to reflect the TRAC-M code.

Verification. Some verification has occurred during the years of TRAC development as documents such as the theory manual[3-6] and adequacy assessment document[3-7] were written or updated, code modifications were undertaken, and code problems were identified and resolved. However, these efforts constitute neither a complete or formal set of verification activities. The last comprehensive review of TRAC by the Advisory Committee on Reactor Safeguards, Reactor Safeguards Subcommittee on Thermal Hydraulic Phenomena was conducted on January 20–21, 1988.

Testing—Validation Using Other Tests. This type of validation of TRAC has taken place, but an expanded set of test problems is envisioned. Problems that test several pieces of coding, test various code features and functions, and evaluate code capabilities via comparison to concept and analytical problems have been employed. A set of such problems is described in Ref. 3-12.

Validation Using Separate Effect Tests. Various SET data have been used throughout the TRAC development history. However, these constitute, at best, a sparse subset of the SET validation (fundamental, component, and several components) needed to fully qualify TRAC-M for its targeted applications. The SET data used as part of the developmental validation of TRAC-M, Version 5.5[3-10, 3-11] are as follows: CCFL using Bankoff data, condensation model using Akimoto's data, critical flow model using Marviken data, core reflood model using Flecht-Seaset, Lehigh and Berkeley tube data, multiple models using UPTF Tests 6 and 8, and CCTF Run 14.

For the last two decades, the majority of validation testing performed for TRAC has used IET data. Although this extensive body of IET validation has shown that TRAC can generally reproduce the major trends and key processes/phenomena for a variety of transients, too little validation of the underlying models and correlations has been performed using SET data.

Testing—Validation Using Integral Effect Tests. As stated in the previous paragraph, numerous validations of various versions of TRAC have been performed using IET data. The majority of these were conducted with TRAC-PF1/MOD1. Because there have been significant changes to the code as it evolved from the MOD1 to the MOD2 version,[3-13] extrapolation of MOD1 assessments to the MOD2 code is problematic. The IET tests used as part of the developmental assessment of TRAC-M, Version 5.5,[3-10, 3-11] are as follows: LOFT L2-6 and L6-1, CCTF Run 54, and SCTF Run 719.

In summary, qualification efforts for the present TRAC-M code constitute a modest fraction of the qualification testing envisioned by NRC's current software quality assurance program and guidelines.[3-14] The validation test matrix, which is defined in subsequent sections of this report, is designed to fulfill the requirements of the NRC guidelines

REFERENCES

3-1. TRAC-PF1/MOD2 Theory Manual, Los Alamos National Laboratory, NUREG/CR-5673, Vol. 1, draft report (July 21, 1993).

3-2. TRAC-BF1/MOD1: An Advanced Best-Estimate Computer Program for BWR Accident Analysis, Idaho National Engineering Laboratory, NUREG/CR-4356, Vol. 1 (August 1992).

3-3. K. E. Carlson et al., RELAP5/MOD3, Code Manual, Idaho National Engineering Laboratory, NUREG/CR-5535, 1994.

3-4. U. S. Rohatgi, et al., RAMONA-4B: A Computer Code with Three-Dimensional Neutron Kinetics for BWR and SBWR System Transients, Brookhaven National Laboratory, NUREG/CR-6359, November 1996.

3-5. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment: Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).

3-6. J. W. Spore, S. J. Jolly-Woodruff, T. K. Knight, J-C. Lin, R. A. Nelson, K. O. Pasamehmetoglu, R. G. Steinke, and C. Unal, "TRAC-PF1/MOD2 Theory Manual," Los Alamos National Laboratory document NUREG/CR-5673, Volume 1 (July 21, 1993).

3-7. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment: Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).

3-8. N. M. Schnurr, R. G. Steinke, V. Martinez, and J. W. Spore, "TRAC-PF1/MOD2 Code Manual: User's Guide," Los Alamos National Laboratory document NUREG/CR-5673, Volume 2 (July 1992).

3-9. L. A. Guffee, S. B. Woodruff, R. G. Steinke, and J. W. Spore, "TRAC-PF1/MOD2 Programmer's Guide," Los Alamos National Laboratory NUREG/CR-5673, Volume 3 (July 1992).

3-10. B. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore and J. L. Steiner, "TRAC-M: Fortran 77, Version 5.5, Developmental Assessment Manual, Volume I: Assessment Sections not including 2D/3D Tests," Los Alamos National Laboratory document LA-UR-99-6480 (December 1999).

3-11. B. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore and J. L. Steiner, "TRAC-M: Fortran 77, Version 5.5, Developmental Assessment Manual, Volume II:

Assessment Sections for 2D/3D Tests," Los Alamos National Laboratory document LA-CP-99-345 (December 1999).

3-12. R. G. Steinke, "A Description of the Test Problems in the TRAC-P Standard Test Matrix," Los Alamos National Laboratory document LA-UR-96-1465 (May 1996).

3-13. P. Giguere, "Comparison of the TRAC-PF1/MOD1 and MOD2 Computer Programs," Los Alamos National Laboratory document LA-UR-96-2220 (1996).

3-14. *Software Quality Assurance Program and Guidelines,* US Nuclear Regulatory Commission document NUREG/BR-0167 (February 1993).

TABLE 3-1
TRAC ELEMENTS BY CATEGORY, SUBCATEGORY, AND MODEL

Category	Subcategory No.	Subcategory Description	Model
BEM	1	Fluid mass equation	Mass convection Mass exchange due to phase change
	2	Fluid momentum equation	Momentum flux Area change Pressure gradient Wall-to-phase momentum exchange Interfacial momentum exchange Momentum exchange due to mass exchange Local losses Gravity
	3	Fluid energy equation	Energy convection Pressure-work term Wall-to-phase energy exchange Interfacial energy exchange Direct energy deposition Energy exchange due to mass exchange
	4	Noncondensable gas and liquid solute	Mass convection Solute mass exchange
	5	3D Vessel model	Refer to the Fluid Mass, Fluid Momentum, Fluid Energy, Noncondensable Gas, and Liquid Solute models.
	6	Heat conduction equation	Lumped-capacitance model 1D radial 2D radial plus axial Reflood implicit Fuel-clad gap Metal-water reaction Material properties

TABLE 3-1 (cont)
TRAC ELEMENTS BY CATEGORY, SUBCATEGORY, AND MODEL

Category	Subcategory No.	Subcategory Description	Model
BEM (continued)	7	Power generation in fuel	Tabular power input Point kinetics 3D kinetics Reactivity feedback Fuel temperature Coolant temperature Void fraction Boron concentration
	8	Radiative energy exchange in the core	Referenced at subcategory level
	9	Equation of state for fluids	Referenced at subcategory level
	10	Fluid thermophysical and transport properties	Referenced at subcategory level
FFEC	1	Regime maps (Also see Appendix C, Table C-1)	Bubbly flow Bubbly slug transition Bubbly slug flow Churn flow Annular-mist flow Transition to stratified flow Stratified flow Plug flow
	2	Fluid mass equation closure (mass exchange)	
	2a	Subcooled boiling (Also see Appendix C, Table C-2)	Referenced at subcategory level
	2b	Interfacial mass exchange (Also see Appendix C, Table C-2)	Referenced at subcategory level
	2c	Solute mass exchange (Also see Appendix C, Table C-2)	Referenced at subcategory level

TABLE 3-1 (cont)
TRAC ELEMENTS BY CATEGORY, SUBCATEGORY, AND MODEL

Category	Subcategory No.	Subcategory Description	Model
FFEC (continued)	3	Fluid momentum equation closure (momentum exchange)	
	3a	Wall-to-phase momentum exchange (Also see Appendix C, Table C-3)	Single phase Two phase, homogeneous Two phase, horizontal stratified
	3b	Interfacial momentum exchange (Also see Appendix C, Table C-4)	Bubbly flow Bubbly slug transition Bubbly slug flow Churn flow Annular-mist flow Transition to stratified flow Stratified flow Plug flow
	3c	Local pressure losses	Abrupt expansion Abrupt contraction Orifice plate User supplied
	4	Fluid energy equation closure (energy exchange)	
	4a	Wall-to-phase energy exchange (Also see Appendix C, Table C-5)	Natural convection to liquid Forced convection to liquid Nucleate boiling Critical heat flux Transition boiling Minimum stable film boiling temperature Film boiling Single-phase vapor Condensation Two-phase forced convection

TABLE 3-1 (cont)
TRAC ELEMENTS BY CATEGORY, SUBCATEGORY, AND MODEL

Category	Subcategory No.	Subcategory Description	Model
FFEC (continued)	4b	Interfacial energy exchange (Also see Appendix C, Table C-6)	Bubbly flow Bubbly slug transition Bubbly slug flow Churn flow Annular-mist flow Transition to stratified flow Stratified flow Plug flow Effect of noncondensables
ECM	1	Centrifugal pumps (Pump component)	Referenced at subcategory level
	2	Steam-water separator (Sepd component)	Referenced at subcategory level
	3	Plenum component	Referenced at subcategory level
	4	Valve component	Referenced at subcategory level
	5	Turbine (Turb component)	Referenced at subcategory level
	6	Pressurizer (Prizer component)	Referenced at subcategory level
SPM	1	Model for countercurrent flow limitation	Referenced at subcategory level
	2	Critical flow model	Referenced at subcategory level
	3	Trip and control elements	Referenced at subcategory level

TABLE 3-1 (cont)
TRAC ELEMENTS BY CATEGORY, SUBCATEGORY, AND MODEL

Category	Subcategory No.	Subcategory Description	Model
SPM (continued)	4	Reflood heat transfer models	
	4a	Flow regime modeling (Also see Appendix C, Table C-3)	Bubbly flow Inverted annular flow Dispersed flow
	4b	Wall-to-phase fluid drag (Also see Appendix C, Table C-3)	Single phase Two phase Homogeneous
	4c	Interfacial fluid drag (Also see Appendix C, Table C-8)	Subcooled boiling Smooth inverted annular flow Rough-wavy inverted annular flow Agitated inverted annular flow Post-agitated (dispersed) flow Highly dispersed flow
	4d	Wall-to-phase fluid heat transfer (Also see Appendix C, Table C-5)	Forced convection to a single-phase liquid Nucleate boiling Critical heat flux Transition boiling Minimum stable film boiling temperature Film boiling Convection to a single-phase vapor Convection to a two-phase mixture Condensation Natural convection to a single-phase liquid
	4e	Interfacial fluid heat transfer (Also see Appendix C, Table C-9)	Bubbly flow Inverted annular flow Dispersed flow
	4f	Conduction heat transfer	Referenced at subcategory level

TABLE 3-1 (cont)
TRAC ELEMENTS BY CATEGORY, SUBCATEGORY, AND MODEL

Category	Subcategory No.	Subcategory Description	Model
SPM (continued)	5	Two-phase level-tracking model	Referenced at subcategory level
	6	Offtake model for Tee component	Referenced at subcategory level
NSM		Fluid field equations	
		1D stability enhancing two-step (SETS) method	Referenced at subcategory level
		3D SETS	Referenced at subcategory level
		Conduction in solid materials	
		1D rectangular and cylindrical	Referenced at subcategory level
		2D rectangular and cylindrical	Referenced at subcategory level
		Power generation in fuel rods	Referenced at subcategory level
		Trip and control system elements	Referenced at subcategory level
		Fluid equation of state	Referenced at subcategory level
		Fluid boundary conditions	Referenced at subcategory level
		Equipment component models	Referenced at subcategory level
		Special-purpose models	Referenced at subcategory level
		Steady-state methods	Referenced at subcategory level
		Timestep size and control methods	Referenced at subcategory level

4.0 PIRT OVERVIEW

Phenomena Identification and Ranking Tables (PIRTs) were first developed during the pioneering Code Scaling, Applicability, and Uncertainty (CSAU) study.[4-1] They have since provided useful support for a number of code-related activities. For the purposes of this report, we focus on the utility of PIRTs in identifying needed code improvements and supporting code development decisions.[4-2]

The purpose of a PIRT is to identify the phenomena that are important to the T-H behavior of a particular plant during a particular transient scenario, e.g., plant event, transient, or accident. In addition, each phenomenon that is deemed of significance is assigned a relative importance ranking, either high, medium, or low, for example. The information obtained through the application of the PIRT process supports the identification of requirements to be imposed on transient T-H codes used to simulate given scenarios.

4.1. PIRT Concepts and Utility

PIRT development proceeds through the following steps:[4-2] (1) specification of the plant design; (2) specification of the scenario(s); (3) establishment of the primary evaluation criteria that will be used to judge the relative importance of phenomena during the scenario; (4) identification, acquisition, and review of all available experimental and analytical data; (5) definition of high-level basic system processes; (6) partitioning of the scenario into characteristic time phases; (7) partitioning the plant design into components; (8) identification of plausible phenomena by phase and component; and (9) ranking component and phenomena importance. Details are provided in Ref. 4-2.

The linkage of the PIRTs and code requirements is evident. First, a given PIRT, i.e., one for a specified plant and scenario, identifies all the components and phenomena that influence the course of the scenario. Second, there is a presumption that all such components and phenomena must be modeled in a transient T-H code used to simulate the scenario so that this information identifies a portion of the code design requirements. Third, some components and phenomena more strongly affect the course of the scenario than others. In fact, some components and phenomena play such a minor role in the progression of the scenario that the course of the scenario is quite insensitive to the details of the component or phenomena. Therefore, the same can be said, about related requirements imposed on the code. The PIRT provides the needed ranking information. Fourth, the ranking information found in a PIRT can also be used as the basis for programmatic decisions about the sequencing of development activities.

A schematic representation of PIRT usage to support development of the Assessment Test Matrix was provided in Fig. 2-3. The PIRT summary discussed in Section 4.3 provides information about phenomena occurring at three levels: local, component, and system. Phenomena occurring at the LL are usually associated with SET data sets (Fig. 2-1), whereas phenomena occurring at the SL are naturally associated with IET data sets. Phenomena occurring at the CL are associated with either SET or IET data sets on a case-by-case basis. Entries in the OST category are most frequently used to test various code features or functions. They are also used to test physical models and the local and CL, although the number of OSTs for this usage is limited.

4.2. PIRT Library

An ideal library would contain PIRTs for each plant type of each U.S. vendor and selected scenarios for each plant type. Unfortunately, such an extensive PIRT library is not available at this time.

The first PIRT was completed in 1989.[4-1] Since that time, a number of additional PIRTs have been completed for PWRs and BWRs; these constitute the current PIRT library for the TRAC-M validation test matrix. The contents of the PWR and BWR PIRT library are identified in the Table 4-1; this table applies only to operational light water reactors within the U.S. A reference to the citation for each PIRT is also provided in Table 4-1.

PIRTs have also been developed for advanced reactors such as the AP600 and the simplified boiling water reactor (SBWR). An AP600 large-break (LB) LOCA PIRT is found in Ref. 4-6. PIRTs for an AP600 SB LOCA, main steam line break (MSLB), and steam SGTR are found in Ref. 4-7. These are not discussed further in this report. PIRTs for SBWR LOCAs are found in Ref. 4-8. Finally, PIRTs have also been developed for other reactor types;[4-2] however, these are not discussed further in this report.

The validation matrix is to cover both PWR and BWR plants, i.e., it is being developed for the consolidated TRAC-M code which has both PWR and BWR capabilities. Given the different design and operating characteristics of PWRs and BWRs, three types of validation tests are envisioned. Tests of the first type are plant-type independent. It is expected, for example, that numerous OSTs and SETs can be used to assess the adequacy of basic models and constitutive relations that are used for both PWR and BWR calculations. Tests of the second type are PWR-specific tests. Tests of the third type are BWR-specific tests. The TRAC validation matrix is an evolutionary validation matrix; the consolidated validation test matrix is expected to evolve with time.

For this release of the matrix documentation, the elements of the PWR validation test matrix are specific to the LB LOCA and SB LOCA applications in Westinghouse plants[4-1,4-3,4-6] and the SB LOCA application in B&W lowered-loop plants.[4-4] Brief descriptions of each PWR and BWR reactor system and scenario included in the PIRT library are provided in Appendix D. The elements of the BWR validation test matrix cover a broader spectrum of events, including the LB LOCA, SB LOCA, and transient events divided into categories based on certain common attributes such as pressurization, depressurization, rapid reactivity increase, coolant temperature decrease, power oscillations, and an ATWS.

Having compiled the individual PWR and BWR PIRT currently available, the next logical step is to develop several summary PIRT tables. The first of these is a PWR summary PIRT. The second is a BWR summary PIRT. Finally, and most importantly, a consolidated PWR and BWR PIRT is developed. The development of these three summary PIRT tables is described in Section 4.3.

TABLE 4-1
PWR AND BWR PIRT LIBRARY

	PWR			BWR[d]		
Category	W[a]	B&W[b]	CE[c]	2	3,4	5,6
Accidents						
LB LOCA	X[4-1,4-6]			X[4-5]	X[4-5]	X[4-5]
SB LOCA	X[4-3]	X[4-4]		X[4-5]	X[4-5]	X[4-5]
SGTR						
MSLB						
ATWS				X[4-5]		
Transients						
Pressurization				X[4-5]		
Depressurization				X[4-5]		
Rapid reactivity increase				X[4-5]		
Coolant temperature decrease				X[4-5]		
Instability	Not Applicable			X[4-5]		

Notes

Number in superscript refer to reference numbers.

a. W plants are further differentiated as 2-loop, 3-loop and 4-loop plants. Additional variations include bundle design (14 x 14, 15 x 15, 16 x1 6, and 17 x 17), number of fuel assemblies and power level (high, medium and low).

b. B&W plants are further differentiated as lowered loop or raised loop. Additional variations include bundle design (15 x 15 and 17 x 17), number of fuel assemblies, and power level (high and low).

c. CE plants are further differentiated on bundle design (14 x 14, 15 x 15 and 16 x 16) and power level (high, low and unique).

d. Individual PIRTs have been produced for BWR/2, BWR/3,4 and BWR/5,6 designs for some accidents as noted, but general BWR PIRTs have been prepared for the ATWS and all the transients.

4.3. Summary Findings for PWR LOCAs

The highly ranked LB LOCA phenomena for W plants are presented in Table 4-2a; this table is based on the PIRTs in Refs. 4-1 and 4-6. The highly ranked SB LOCA phenomena for W plants are presented in Table 4-2b; this table is based on the PIRT in Ref. 4-3. The highly ranked SB LOCA phenomena for B&W lowered-loop plants are presented in Table 4-2c; this table is based on the PIRT in Ref. 4-4.

Our summary of highly ranked PWR LOCA phenomena is presented as Table 4-2d. This table summarizes highly ranked phenomena from Refs. 4-1, 4-3, 4-4 and 4-6; identifies whether the phenomena is evident at the LL, CL, SL, or in multiple levels; and identifies the associated TRAC models as organized and discussed in Section 3.

In previous efforts to prepare a summary PIRT for all PWR phenomena,[4-9] we encountered and addressed several issues. First, different phenomena names were used in the individual PIRTs to describe identical phenomena. For our summary tabulation, we selected a unique and consistent set of phenomena names and recast the individual PIRTs using this set of phenomena names. Our definitions for the highly ranked PWR LB LOCA PIRT phenomena identifiers in Table 4-2a-c and the summary tabulation of highly ranked PWR LOCA phenomena are provided in Table 4-3. In addition, Table 4-3 contains the definitions of the highly ranked BWR phenomena discussed in the next section.

PWR PIRTs have been developed for only LOCAs to date. They have not been developed for either non-LOCA accidents or transient sequences.

4.4. Summary Findings for BWR Events

Highly ranked LB LOCA phenomena for BWR plants are presented in Table 4-4a; this table is based on the PIRTs in Refs. 4-5. Highly ranked SB LOCA phenomena for BWR plants are presented in Table 4-4b; this table is also based on the PIRTs in Ref. 4-5. For the LB LOCA (Table 4-4a) and SB LOCA (Table 4-4b), the PIRTs have been developed for the following three types of BWRs: (1) BWR/2, (2) BWR/3 and /4, and (3) BWR/5 and /6. Highly ranked phenomena for BWR transients are presented in Table 4-4c, also based on the PIRTs in Ref. 4-5. The transient event categories covered are pressurization, depressurization, rapid reactivity increase, coolant temperature decrease, instability (power oscillation), and ATWS.

Our summary of highly ranked BWR phenomena is presented in Table 4-2d. This table summarizes highly ranked phenomena for the spectrum of PIRT scenarios presented in Ref. 4-5; identifies whether the phenomena is occurs at the LL, CL, SL; and identifies the associated TRAC models as organized and discussed in Section 3.

Our definitions for the highly ranked BWR PIRT phenomena identifiers in Table 4-4d are provided in Table 4-3.

4.5. Summary Findings for PWR and BWR Events

Finally, the summary PWR PIRT findings (Table 4-2d) and summary BWR PIRT findings (Table 4-4d) have been consolidated into a single table of highly ranked light water reactor phenomena (Table 4-5) for which PIRTs are available. We do note that PIRTs do not exist for all PWR plant types and accident sequence. Nevertheless, the list in Table 4-5 is believed to represent the majority of the highly important T-H processes occurring in light water reactors. The list can be easily updated as addition PIRTs are generated for other PWRs and accident sequences.

4.6. Application to TRAC-M Qualification

Table 4-2d lists the highly ranked phenomena for the PWR LOCAs. Table 4-4d lists the high-ranked phenomena for the BWR events described in Section 4.2. TRAC must model these phenomena. The phenomena identified in Tables 4-2d and 4-4d occur at different levels within a plant or facility. There is a natural association between LL phenomena and the flow field models and engineering correlations FFEC described in Section 3.2 and the SPM and associated tables described in Section 3.4. The appropriate cross-correlation or linkage between phenomena identified in the summary PIRT tabulation and the associated models for highly ranked phenomena in PWRs is provided in Table 4-2a-c. The appropriate cross-correlation or linkage between phenomena identified in the summary PIRT tabulation and the associated models for highly ranked phenomena in BWRs is provided in Table 4-4a-c.

There are two possible associations between CL phenomena and TRAC models. For some CL phenomena, there is no unique TRAC component model. Thus, the modeling capability is founded in more fundamental TRAC components and the underlying flow FFEC. For other CL phenomena, specific TRAC component models do exist, e.g., the Pump.

Some of the phenomena listed in Tables 4-2 and 4-4, are SL phenomena. These phenomena can invoke the entire hierarchy of TRAC models; basic equation models, as described in Section 3.1; flow field models and engineering correlations, as described in Section 3.2; equipment component models, as described in Section 3.3; and special-purpose models, as described in Section 3.4.

In summary, the cross–correlation of TRAC-M models at all levels, i.e., local, component, and system, with the summary PIRT phenomena and component lists serve to identify the associated TRAC models that must be provided and qualified.

REFERENCES

4-1. Technical Program Group, EG&G Idaho, Inc., Quantifying Reactor Safety Margins: Application of CSAU to a LB LOCA, USNRC report NUREG/CR-5249, 1989.

4-2. G. E. Wilson and B. E. Boyack, “The Role of the PIRT Process in Experiments, Code Development and Code Application Associated with Reactor Safety Analysis,” Nuclear Engineering and Design, Vol. 186 (1998).

4-3. S. M. Bajorek, A. Ginsberg, D. J. Shimeck, K. Ohkawa, M. Y. Young, L. E. Hochreiter, P. Griffith, Y. Hassan, T. Fernandez, and D. Speyer, “Small Break Loss of Coolant Accident Phenomena Identification and Ranking Table (PIRT) for Westinghouse Pressurized Water Reactors,” Proceedings of the Ninth International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-9), San Francisco, California (October 3–8, 1999).

4-4. M. G. Ortiz, “Uncertainty Analysis of Minimum Vessel Liquid Inventory during a Small Break LOCA in a B&W Plant: An Application of the CSAU Methodology,” Idaho National Engineering Laboratory document NUREG/CR-5818 (December 1992).

4-5. M. Straka and L. W. Ward, “BWR PIRT and Assessment Matrices for BWR LOCA and Non-LOCA Events,” Scientech, Inc. document SCIE-NRC-393-99 (1999).

4-6. B. E. Boyack, “AP600 LB LOCA Phenomena Identification and Ranking Tabulation,” Los Alamos National Laboratory document LA-UR-95-2718 (1995).

4-7. J. D. Burtt, C. D. Fletcher, G. E. Wilson, C. B. Davis, and T. J. Boucher, “Phenomena Identification and Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios,” Idaho National Engineering Laboratory document INEL-94/0061, Rev. 2 (November 1996).

4-8. P. G. Kroeger et al., “Preliminary Phenomena Identification and Ranking Tables for SBWR LOCA Scenarios,” Brookhaven National Laboratory Technical letter report W-6092-5 6/95 (June 1995).

4-9. E. D. Hughes and B. E. Boyack, “TRAC-P Validation Test Matrix,” Los Alamos National Laboratory document LA-UR-97-3900 (September 5, 1997).

TABLE 4-2a
SUMMARY TABULATION OF HIGHLY RANKED W PWR LB LOCA PHENOMENA[a]

Phenomena	Ref.[a]	Level	Phase[b]	TRAC Models[c] (category: subcategory: model)
Asymmetries	4-1	SL	1, 2	BEM:all:fluid flow equations
Boiling—film	4-1, 4-6	LL	1, 2, 3	FFEC:4a: film boiling
Boiling—transition	4-1, 4-6	LL	1, 2, 3	FFEC:4a:transition boiling
Condensation—interfacial	4-1	LL	2	FFEC:4b:all flow regimes
Draining	4-6	LL	4	BEM:all:fluid flow equations FFEC:3:all flow regimes
Entrainment/deentrainment	4-1	LL	2, 3	FFEC:3b:all flow regimes
Evaporation—interfacial	4-1, 4-6	LL	1, 2, 3	FFEC:4a:all flow regimes
Flashing—interfacial	4-1, 4-6	LL	1	FFEC:4b:all flow regimes
Flow—countercurrent	4-1	CL	2	FFEC:3b:all flow regimes
Flow—critical	4-1, 4-6	LL	1, 2	SPM:2:critical flow model
Flow—discharge	4-6	LL	2, 3	BEM:all:fluid flow equations FFEC:3:all flow regimes
Flow—multidimensional	4-1, 4-6	CL	2, 3	BEM:5:3D vessel model
Heat conductance—fuel-clad gap	4-1, 4-6	LL	1	BEM:6:fuel-clad gap model
Heat transfer—forced convection to vapor	4-1, 4-6	LL	2	FFEC:4a:single phase vapor
Heat transfer—stored energy release	4-1, 4-6	LL	1	BEM:6:conduction equation, fuel-clad gap
Interfacial shear	4-1, 4-6	LL	2, 3	FFEC:3b:all flow regimes
Level	4-1, 4-6	SL	3	BEM:all:fluid flow equations
Noncondensable effects	4-1	LL	3	FFEC:4b:effect of noncondensables
Oscillations	4-1, 4-6	SL,CL	3	BEM:all:fluid flow equations FFEC:3: all flow regimes FFEC:4: all flow regimes
Power—decay heat	4-1, 4-6	CL	2, 3, 4	BEM:7:power generation in fuel
Pump—performance, inc. degradation	4-1, 4-6	CL	1	ECM:1:centrifugal pump component
Reactivity—void	4-6	CL	1	BEM:7:power generation, reactivity feedback

[a] Based on Westinghouse 4-loop plant of CSAU study (Ref. 4-1) and AP600 plant (Ref. 4-6).
[b] Phase of the LB LOCA sequence: Blowdown = 1, Refill = 2, Reflood = 3, Long-Term = 4
[c] Per Section 3, there are five model categories. Each model category has subentries: BEM is Basic Equation Model, FFEC is Flow Field Model and Engineering Correlation (Closure), ECM is Equipment Component Model, SPM is Special Purpose Model, and NSM is Numerical Special Model.

TABLE 4-2b
SUMMARY TABULATION OF HIGHLY RANKED W PWR SB LOCA PHENOMENA[a]

Phenomena	Ref.[a]	Level	Phase[b]	TRAC Models[c] (category: subcategory: model)
Condensation—fluid to surface	4-3	LL	1,3	FFEC:4a:condensation
Condensation—interfacial	4-3	LL	4,5	FFEC:4b:all flow regimes
Entrainment / deentrainment	4-3	LL	3	FFEC:3b:all flow regimes
Flashing—interfacial	4-3	LL	3,4,5	FFEC:4b:all flow regimes
Flow regime—break inlet	4-3	CL	all	FFEC:1:all flow regimes
Flow—countercurrent	4-3	CL	2,3	FFEC:3b:all flow regimes
Flow—critical	4-3	LL	all	SPM:2:critical flow model
Flow—gap	4-3	CL	3	BEM:all:fluid flow equations
Heat Transfer—post-CHF	4-3	LL	4,5	FFEC:4a, 4b;transition boiling, film boiling
Interfacial shear	4-3	LL	3	FFEC:3b:all flow regimes
Level	4-3	SL	3,4,5	BEM:all:fluid flow equations
Oxidation	4-3	LL	4,5	BEM:6:metal-water reaction
Power—3D distribution	4-3	CL	4,5	BEM:7:3D kinetics
Power—decay heat	4-3	CL	all	BEM:7:power generation in fuel
Power—local peaking (fuel rod)	4-3	CL	4,5	BEM:7:3D kinetics
Pressure drop	4-3	CL	3	BEM:all:fluid flow equations FFEC:3,4;all
Rewet	4-3	LL	4,5	FFEC:4a SPM:4d
Stratification—horizontal	4-3	CL	3	BEM:1,2,3 FFEC:1:stratified flow

[a] Based on Westinghouse 4-loop plant; stated by PIRT panel to have extended applicability to conventional Westinghouse 3- and 4-loop plants (Ref. 4-3).
[b] Phase of the SB LOCA sequence: Blowdown = 1, Natural Circulation = 2, Loop Seal Clearance = 3, Boil-off = 4, and Core Recovery = 5.
[c] Per Section 3, there are five model categories. Each model category has subentries: BEM is Basic Equation Model, FFEC is Flow Field Model and Engineering Correlation (Closure), ECM is Equipment Component Model, SPM is Special Purpose Model, and NSM is Numerical Special Model.

TABLE 4-2c
SUMMARY TABULATION OF HIGHLY RANKED B&W PWR SB LOCA PHENOMENA[a]

Phenomena	Ref.[a]	Level	Phase[b]	TRAC Models[c] (category: subcategory: model)
Flow—critical	4-4	LL	1,2,4	SPM:2:critical flow model
Flow—high pressure injection	4-4	LL	3,4	BEM:all fluid flow equations FFEC:all
Flow—natural circulation	4-4	SL	2	BEM:all:fluid flow equations
Heat Transfer—primary to secondary	4-4	LL	4	BEM:all fluid flow equations BEM:6:1D radial FFEC:all
Level	4-4	SL	2	BEM:all:fluid flow equations
Power—decay heat	4-4	CL	2	BEM:7:power generation in fuel
Pump—performance, inc. degradation	4-4	CL	3	ECM:1:centrifugal pump component

[a] Based on Babcock & Wilcox 2x4-loop, lowered-loop plant (Ref. 4-4).
[b] Phase of the SB LOCA sequence: Blowdown = 1, Natural Circulation = 2, Loss of Natural Circulation = 3, and Boiler-Condenser = 4.
[c] Per Section 3, there are five model categories. Each model category has subentries: BEM is Basic Equation Model, FFEC is Flow Field Model and Engineering Correlation (Closure), ECM is Equipment Component Model, SPM is Special Purpose Model, and NSM is Numerical Special Model.

TABLE 4-2d
SUMMARY TABULATION OF HIGHLY RANKED PWR LOCA PHENOMENA

Phenomena	Level	Event Type: W LB LOCA	Event Type: W SB LOCA	Event Type: B&W SB LOCA
Boiling—film	LL	X		
Boiling—transition	LL	X		
Condensation—fluid to surface	LL		X	
Condensation—interfacial	LL	X	X	
Draining	LL	X		
Entrainment / deentrainment	LL	X	X	
Evaporation—interfacial	LL	X		
Flashing—interfacial	LL	X	X	
Flow—critical	LL	X	X	X
Flow—discharge	LL	X		
Flow—high pressure injection	LL			X
Heat conductance—fuel-clad gap	LL	X		
Heat transfer—forced convection to vapor	LL	X		
Heat transfer—post-CHF	LL		X	
Heat transfer—primary to secondary	LL			X
Heat transfer—stored energy release	LL	X		
Interfacial shear	LL	X	X	
Noncondensable effects	LL	X		
Oxidation	LL		X	
Rewet	LL		X	
Flow regime—break inlet	CL		X	
Flow—countercurrent	CL	X	X	
Flow—gap	CL		X	
Flow—multidimensional	CL	X		
Oscillations	CL	X		
Power—3D distribution	CL		X	
Power—decay heat	CL	X	X	X
Power—local peaking (fuel rod)	CL		X	
Pressure drop	CL		X	
Pump—performance, inc. degradation	CL	X		X
Reactivity—void	CL	X		
Stratification—horizontal	CL		X	
Asymmetries	SL	X		
Flow—natural circulation	SL			X
Level	SL	X	X	X
Oscillations	SL	X		

TABLE 4-3
CONSOLIDATED PIRT PHENOMENA DESCRIPTIONS[a]

PIRT Term	Description
Asymmetries	A difference in T-H behavior that can be attributed to the geometrically asymmetric arrangement of hardware.
Boiling—film	Boiling regime in which vapor blankets all or an appreciable portion of the heating surface.
Boiling—nucleate	A boiling regime in which bubble formation is at the liquid-solid interface which results in slow surface temperature increases for relatively large in creases in surface heat flux.
Boiling—transition	A boiling regime that spans the boiling surface between critical heat flux and minimum film boiling.
Boiling—subcooled	A boiling regime in beginning with the onset of nucleate boiling and continuing to the onset of saturated boiling, the boundary between the latter two regimes occurring when the bulk liquid temperature approaches saturation at the given pressure.
Condensation—fluid to surface	The process whereby steam is cooled due to contact with a colder surface, resulting in a change of phase from vapor to liquid at the surface.
Condensation—interfacial	The process whereby steam is cooled due to contact with a colder liquid, resulting in a change of phase from vapor to liquid at the interface between the two phases.
Draining	The downward flow of fluid on a surface under the influence of gravity.
Dryout-critical heat flux	Also variously called burnout, boiling crisis, and critical heat flux. The point in a heated channel with flowing two-phase flow at which there is no longer any liquid in contact with the heated surface, resulting in a rapid increase in surface temperature.

[a] If available, the descriptions are taken from Ref. 4-6. Additional terms are based on definitions found in the *Dictionary of Scientific and Technical Terms,* 2nd edition, McGraw-Hill Book Company (1978).

TABLE 4-3 (cont)
PWR PIRT PHENOMENA DESCRIPTIONS

PIRT Term	Description
Entrainment/deentrainment	The process whereby liquid is captured (entrained) by a high-velocity steam flow. The process whereby liquid departs (deentrained) from a steam flow.
Evaporation—interfacial	The process whereby a fluid changes from the liquid state to the vapor state by the addition of energy.
Flashing—interfacial	The process whereby fluid changes from the liquid state to the vapor state due to a reduction in the fluid pressure, which lowers the saturation temperature.
Flow regime—break inlet	The characteristics of the flow at the break entrance, e.g., subcooled liquid, saturated, two-phase, stratified, vapor, etc.
Flow—carryunder	The mass fraction of produced steam that is entrained via the separator liquid drain path.
Flow—countercurrent	The process whereby liquid flows opposite (counter) to the gas flow direction.
Flow—channel-bypass leakage	Flow via the channel-bypass leakage path.
Flow—critical	The maximum possible flow through a flow constricting item of hardware, usually a nozzle, orifice, or break in a pipe.
Flow—discharge	Flow leaving a component under the influence of an upstream forcing function.
Flow—distribution	The location of fluid (liquid and vapor) throughout a system
Flow—forward (jet pumps)	That part of the jet pump operating regime in which the outlet (discharge) flow is positive, i.e. forward.
Flow—gap	Flow through the hot leg to downcomer gap.
Flow—multidimensional	Flow that has two or more dominant velocity vectors. Examples are multidimensional flows in a PWR core during reflooding and spray induced flows in the upper plenum of a BWR.

TABLE 4-3 (cont)
PWR PIRT PHENOMENA DESCRIPTIONS

PIRT Term	Description
Flow—multi-channel T/H effect	Differences in the boiling-induced flows and pressure drop characteristics in parallel channels, e.g., fuel assemblies that may induce dynamic instabilities.
Flow—reverse (jet pumps)	That part of the jet pump operating regime in which the outlet flow is negative, i.e. reversed.
Heat conductance—fuel-clad gap	The overall thermal resistance to the flow of heat between the fuel pellets and cladding in a nuclear fuel rod.
Heat conductance—fuel	The overall thermal resistance to the flow of heat from the high temperature to lower-temperature parts of the fuel pellet.
Heat—stored	The total energy residing in a material at a given time; the amount being dependent on the material mass, heat capacity and temperature.
Heat transfer—forced convection to vapor	Process of energy transport by the combined action of heat conduction, energy storage, and mixing motion.
Heat transfer—post CHF	Heat transfer between the two-phase fluid and the heated surface in the liquid-deficient region downstream of the CHF point, i.e., the location at which the heat transfer condition of the two-phase flow substantially deteriorates.
Heat transfer—radiation	The transfer of energy from a higher temperature body to a lower temperature body without relying on the intervening medium, i.e., the transfer can take place in a vacuum.
Heat transfer—stored energy release	The process by which the energy within a solid structure is released to a lower energy state through one or more heat transfer processes, e.g., conduction and convection. Applies specifically to the transport of the energy residing in fuel rods operating at full power to the coolant following a reactor trip.
Interfacial shear	The friction caused by the velocity difference between two phases at their interface.
Level	The vertical height of a column of single- or two-phase fluid.

TABLE 4-3 (cont)
PWR PIRT PHENOMENA DESCRIPTIONS

PIRT Term	Description
Noncondensable effects	The impact of the presence of noncondensable gases upon heat transfer or any other phenomenon such as flow, condensation, flashing, and vapor volume expansion.
Oscillations	The periodic variation of any given hydraulic characteristic between two values.
Oxidation	A chemical reaction that increases the oxidation content of a material. Of specific interest is cladding oxidation, which occurs at elevated temperatures, which can occur only under accident conditions.
Power—3D distribution	The axial, radial and azimuthal power variation in a core.
Power—3D kinetics effect	Neutronic effect that takes place in space, i.e. three dimensions.
Power—decay heat	Heat produced by the decay of radioactive nuclides.
Power—local peaking (fuel rod)	The ratio of power at a location (specific fuel rod) to the core average power.
Pressure drop	The reduction in pressure with distance.
Pressure wave propagation	The movement of a compression or decompression wave through the coolant.
Pump—performance, including degradation	The behavior of a pump under all normal and off-normal conditions.
Reactivity—fuel temperature	Prompt reactivity feedback from fuel temperature changes, also known as Doppler feedback.
Reactivity—scram	Reactor trip initiates insertion of control rods and their associated negative reactivity into the core.
Reactivity—void	The change in core reactivity due to an increase or decrease in the amount of void in the moderating fluid.
Rewet	The post-dryout process in which liquid once again resumes intimate contact with a heated surface.

TABLE 4-3 (cont)
PWR PIRT PHENOMENA DESCRIPTIONS

PIRT Term	Description
Spray distribution	The radial and azimuthal distribution of flow in the upper plenum resulting from operation of the spray system.
Stability—neutronic and T/H interaction	Neutronic-T-H interaction between fuel channel boiling and nuclear reactivity feedback processes.
Stratification—horizontal	The variation of physical properties such as temperature or density across the vertical cross section of a fluid body having a primarily horizontal orientation, e.g., the cold leg of a nuclear steam supply system.
Subcooling—coolant	The difference between the saturation temperature at a given pressure and the temperature of the coolant. The degree of subcooling affects density-wave travel time and two-phase pressure drop via boiling boundary change.
Void collapse	The rapid reduction in void in the core.
Void distribution	The distribution (location) of two-phase fluid within the nuclear steam supply system.

TABLE 4-4a
SUMMARY TABULATION OF HIGHLY RANKED BWR LB LOCA PHENOMENA[a]

Phenomena	Ref.[a]	Level	Phase[b]	TRAC Models[c] (category: subcategory: model)
Boiling—film	4-5	LL	1,2,3	FFEC:4a: film boiling
Boiling—nucleate	4-5	LL	4	FFEC:4a:nucleate boiling
Condensation—interfacial	4-5	LL	1,2,3	FFEC:4b:all flow regimes
Dryout–critical heat flux	4-5	LL	1,2,3	FFEC:4a:critical heat flux
Flashing—interfacial	4-5	LL	1	FFEC:4b:all flow regimes
Flow—channel-bypass leakage	4-5	CL	1,2,3	FFEC:3b:all flow regimes
Flow—countercurrent	4-5	CL	1,2,3	FFEC:3b:all flow regimes
Flow—critical	4-5	LL	1	SPM:2:critical flow model
Flow—distribution	4-5	CL	1	FFEC:3b:all flow regimes
Flow—forward (jet pumps)	4-5	CL	1	FFEC:3b:all flow regimes
Flow—multidimensional	4-5	CL	1,2,3,4	BEM:5:3D vessel model
Flow—natural circulation	4-5	SL	2,3,4	BEM:all:fluid flow equations
Flow—reverse (jet pumps)	4-5	CL	1	FFEC:3b:all flow regimes
Heat transfer–fuel-clad gap	4-5	LL	1	BEM:6:fuel-clad gap model
Heat transfer—forced convection to vapor	4-5	LL	2,3	FFEC:4a:single phase vapor
Heat transfer—radiation	4-5	LL	2,3	BEM:8:radiative energy exchange in the core
Heat—stored	4-5	LL	1,2,3	BEM:6:material properties
Interfacial shear	4-5	LL	1,2,3	FFEC:3b:all flow regimes
Level	4-5	SL	1,2,3,4	BEM:all:fluid flow equations
Power—3D distribution	4-5	CL	2,3	BEM:7:3D kinetics
Power—decay heat	4-5	CL	1,2,3,4	BEM:7:power generation in fuel
Pressure drop	4-5	CL	1	BEM:all:fluid flow equations FFEC:3,4;all
Pump performance, inc. degradation	4-5	CL	1	ECM:1:centrifugal pump component
Rewet	4-5	LL	2,3,4	FFEC:4a SPM:4d
Spray distribution	4-5	CL	1,2,3,4	BEM:all:fluid flow equations, FFEC 4
Void distribution	4-5	CL	1,2,3,4	BEM:all:fluid flow equations FFEC:all SPM:4

[a] Based on BWR/2, BWR/3 and 4, and BWR/5 and 6 designs as discussed in Ref. 4-5.
[b] Phase of the LB LOCA sequence: Blowdown = 1, Refill = 2, Reflood = 3, Long-Term = 4.
[c] Per Section 3, there are five model categories. Each model category has subentries: BEM is Basic Equation Model, FFEC is Flow Field Model and Engineering Correlation (Closure), ECM is Equipment Component Model, SPM is Special Purpose Model, and NSM is Numerical Special Model.

TABLE 4-4b
SUMMARY TABULATION OF HIGHLY RANKED BWR SB LOCA PHENOMENA[a]

Phenomena	Ref.[a]	Level	Phase[b]	TRAC Models[c] (category: subcategory: model)
Boiling—film	4-5	LL	b,2,3	FFEC:4a: film boiling
Boiling—nucleate	4-5	LL	a,4	FFEC:4a:nucleate boiling
Condensation—interfacial	4-5	LL	b,2,3,4	FFEC:4b:all flow regimes
Dryout–critical heat flux	4-5	LL	b,3,4	FFEC:4a:critical heat flux
Flashing—interfacial	4-5	LL	b	FFEC:4b:all flow regimes
Flow—channel-bypass leakage	4-5	CL	b,3,4	FFEC:3b:all flow regimes
Flow—countercurrent	4-5	CL	b,3,4	FFEC:3b:all flow regimes
Flow—critical	4-5	LL	a,b	SPM:2:critical flow model
Flow—distribution	4-5	CL	b	FFEC:3b:all flow regimes
Flow—forward (jet pumps)	4-5	CL	a,b	FFEC:3b:all flow regimes
Flow—multidimensional	4-5	CL	b,2,3,4	BEM:5:3D vessel model
Flow—natural circulation	4-5	SL	b,2,3,4	BEM:all:fluid flow equations
Flow—reverse (jet pumps)	4-5	CL	b	BEM:all:fluid flow equations
Heat transfer–fuel-clad gap	4-5	LL	b	BEM:6:fuel-clad gap model
Heat transfer—forced convection to vapor	4-5	LL	3,4	FFEC:4a:single phase vapor
Heat—stored	4-5	LL	b,2,3	BEM:6:material properties
Interfacial shear	4-5	LL	a,b,2,3,4	FFEC:3b:all flow regimes
Level	4-5	SL	b,2,3,4	BEM:all:fluid flow equations
Power—3D distribution	4-5	CL	2,3	BEM:7:3D kinetics
Power—decay heat	4-5	CL	a,b,2,3,4	BEM:7:power generation in fuel
Pressure drop	4-5	CL	a,b	BEM:all:fluid flow equations FFEC:3,4;all
Pump–performance, inc. degradation	4-5	CL	a	ECM:1:centrifugal pump component
Reactivity—scram	4-5	SL	a	BEM:7
Rewet	4-5	LL	b,2,3	FFEC:4a SPM:4d
Spray distribution	4-5	CL	b,2,3,4	BEM:all:fluid flow equations, FFEC 4
Void distribution	4-5	CL	a,b,2,3	BEM:all:fluid flow equations FFEC:all SPM:4

[a] Based on BWR/2, BWR/3 and 4, and BWR/5 and 6 designs as discussed in Ref. 4-5.
[b] Phase of the LB LOCA sequence: Blowdown before ADS operation = a, Blowdown after ADS operation = b, Refill = 2, Reflood = 3, Long-Term = 4.
[c] Per Section 3, there are five model categories. Each model category has subentries: BEM is Basic Equation Model, FFEC is Flow Field Model and Engineering Correlation (Closure), ECM is Equipment Component Model, SPM is Special Purpose Model, and NSM is Numerical Special Model.

TABLE 4-4c
SUMMARY TABULATION OF HIGHLY RANKED BWR TRANSIENT PHENOMENA[a]

Phenomena	Ref.[a]	Level	ransient[b]	TRAC Models[c] (category: subcategory: model)
Boiling—film	4-5	LL	3,4,5	FFEC:4a: film boiling
Boiling—subcooled	4-5	LL	5	FFEC:4a:nucleate boiling
Condensation—interfacial	4-5	LL	4	FFEC:4b:all flow regimes
Dryout–critical heat flux	4-5	LL	3,4,5	FFEC:4a:critical heat flux
Flow—carry-under	4-5	SL	1,2,4,5,6	BEM:all:fluid flow equations
Flow—critical	4-5	LL	1,2,6	SPM:2:critical flow model
Flow—forward (jet pumps)	4-5	CL	1,2,4,5,6	FFEC:3b:all flow regimes
Flow—multi-channel T/H effect	4-5	CL	all	BEM:all:fluid flow equations
Flow—multidimensional	4-5	CL	4,5	BEM:5:3D vessel model
Flow—natural circulation	4-5	SL	5	BEM:all:fluid flow equations
Heat conductance—fuel-clad gap	4-5	LL	1,3,5,6	BEM:6:fuel-clad gap model
Interfacial shear	4-5	LL	all	FFEC:3b:all flow regimes
Level	4-5	SL	1,2,4,5,6	BEM:all:fluid flow equations
Power—3D distribution	4-5	CL	3,5	BEM:7:3D kinetics
Power—3D kinetics effect	4-5	CL	1,3,4,5,6	BEM:7:3D kinetics
Pressure drop	4-5	CL	all	BEM:all:fluid flow equations FFEC:3,4;all
Pressure wave propagation	4-5	SL	1,2,6	BEM:all:fluid flow equations
Pump–performance, inc. degradation	4-5	CL	5,6	ECM:1:centrifugal pump component
Reactivity—fuel temperature	4-5	CL	1,3,4,5,6	BEM:7:power generation, reactivity feedback
Reactivity—scram	4-5	SL	1,5,6	BEM:7
Reactivity—void	4-5	CL	All	BEM:7:power generation, reactivity feedback
Stability—neutronic and T/H interaction	4-5	SL	5	BEM:all:fluid flow equations BEM:7:power generation, reactivity feedback
Subcooling—coolant	4-5	SL	5	BEM:all:fluid flow equations
Void collapse	4-5	CL	1,3,4,6	BEM:all:fluid flow equations, FFEC 4
Void distribution	4-5	CL	all	BEM:all:fluid flow equations FFEC:all SPM:4
Void—subcooled liquid	4-5	CL	all	FFEC:2a:subcooled boiling

[a] Based on BWR/2, BWR/3 and 4, and BWR/5 and 6 designs as discussed in Ref. 4-5.
[b] Transients are pressurization = 1, depressurization = 2, rapid reactivity increase = 3, coolant temperature decrease = 4, instability (power oscillations) = 5 and anticipated transient without scram (ATWS) = 6.
[c] Per Section 3, there are five model categories. Each model category has subentries: BEM is Basic Equation Model, FFEC is Flow Field Model and Engineering Correlation (Closure), ECM is Equipment Component Model, SPM is Special Purpose Model, and NSM is Numerical Special Model.

TABLE 4-4d
SUMMARY TABULATION OF HIGHLY RANKED BWR PHENOMENA

Phenomena	Level	Event Type		
		LB LOCA	SB LOCA	Transient
Boiling—film	LL	X	X	X
Boiling—nucleate	LL	X	X	
Boiling—subcooled	LL			X
Condensation—interfacial	LL	X	X	X
Dryout–critical heat flux	LL	X	X	X
Flashing—interfacial	LL	X	X	
Flow—critical	LL	X	X	X
Heat conductance—fuel-clad gap	LL	X	X	X
Heat transfer—forced convection to vapor	LL	X	X	
Heat transfer—radiation	LL	X		
Heat—stored	LL	X	X	
Interfacial shear	LL	X	X	X
Rewet	LL	X	X	
Flow—channel-bypass leakage	CL	X	X	
Flow—countercurrent	CL	X	X	
Flow—distribution	CL	X	X	
Flow—forward (jet pumps)	CL	X	X	X
Flow—multi-channel T/H effect	CL			X
Flow—Multidimensional	CL	X	X	X
Flow—reverse (jet pumps)	CL	X	X	
Power—3D distribution	CL	X	X	X
Power—3D kinetics effect	CL			X
Power—decay heat	CL	X	X	
Pressure drop	CL	X	X	X
Pump–performance, inc. degradation	CL	X	X	X
Reactivity—fuel temperature	CL			X
Reactivity—void	CL			X
Spray distribution	CL	X	X	
Void collapse	CL			X
Void distribution	CL	X	X	X
Void—subcooled liquid	CL			X
Flow—carry-under	SL			X
Flow—natural circulation	SL	X	X	X
Level	SL	X	X	X
Pressure wave propagation	SL			X
Reactivity—scram	SL		X	X
Stability—neutronic and T/H interaction	SL			X
Subcooling—coolant	SL			X

TABLE 4-5
CONSOLIDATED TABULATION OF HIGHLY RANKED PIRT PHENOMENA

		Transient Type					
Phenomena	Level	W-P LB LOCA	W-P SB LOCA	B&W-P SB LOCA	GE-BWR LB LOCA	GE-BWR SB LOCA	GE-BWR TRANSIENT
Boiling—film	LL	X			X	X	X
Boiling—nucleate	LL				X	X	
Boiling—subcooled	LL						X
Boiling—transition	LL	X					
Condensation—fluid to surface	LL		X				
Condensation—interfacial	LL	X	X		X	X	X
Draining	LL	X					
Dryout–critical heat flux	LL				X	X	X
Entrainment/deentrainment	LL	X	X				
Evaporation—interfacial	LL	X					
Flashing—interfacial	LL	X	X		X	X	
Flow—critical	LL	X	X	X	X	X	X
Flow—discharge	LL	X					
Flow—high pressure injection	LL			X			
Heat conductance—fuel-clad gap	LL	X			X	X	X
Heat transfer—forced convection to vapor	LL	X			X	X	
Heat Transfer—post-CHF	LL		X				
Heat Transfer—primary to secondary	LL			X			
Heat transfer—radiation	LL				X		
Heat transfer—stored energy release	LL	X					
Heat—stored	LL				X	X	
Interfacial shear	LL	X	X		X	X	X
Noncondensable effects	LL	X					
Oxidation	LL		X				
Rewet	LL		X		X	X	
Flow—channel-bypass leakage	CL				X	X	
Flow—countercurrent	CL	X	X		X	X	
Flow—distribution	CL				X	X	
Flow—forward (jet pumps)	CL				X	X	X

TABLE 4-5 (cont)
CONSOLIDATED TABULATION OF HIGHLY RANKED PIRT PHENOMENA

		Transient Type					
Phenomena	Level	W-P LB LOCA	W-P SB LOCA	B&W-P SB LOCA	GE-BWR LB LOCA	GE-BWR SB LOCA	GE-BWR TRANSIENT
Flow regime—break inlet	CL		X				
Flow—gap	CL		X				
Flow—multi-channel T/H effect	CL						X
Flow—multidimensional	CL	X			X	X	X
Flow—reverse (jet pumps)	CL				X	X	
Oscillations	CL	X					
Power—3D distribution	CL		X		X	X	X
Power—3D kinetics effect	CL						X
Power—decay heat	CL	X	X	X	X	X	
Power—local peaking (fuel rod)	CL		X				
Pressure drop	CL		X		X	X	X
Pump[a]—performance, inc. degradation	CL	X		X	X	X	X
Reactivity—fuel temperature	CL						X
Reactivity—void	CL	X					X
Spray distribution	CL				X	X	
Stratification—horizontal	CL		X				
Void collapse	CL						X
Void distribution	CL				X	X	X
Void—subcooled liquid	CL						X
Asymmetries	SL	X					
Flow—carry-under	SL						X
Flow—natural circulation	SL			X	X	X	X
Level	SL	X	X	X	X	X	X
Oscillations	SL	X					
Pressure wave propagation	SL						X
Reactivity—scram	SL					X	X
Stability—neutronic and T/H interaction	SL						X
Subcooling—coolant	SL						X

[a]centrifugal.

5.0. PLANT TYPES AND TARGETED APPLICATIONS

T-H codes are specifically designed for a variety of targeted applications. Among these applications are (1) reactor safety analyses for both operating and planned reactors, (2) audits of licensee's calculations, (3) analyses of operating reactor events, (4) analyses of accident management strategies, (5) support for test planning and interpretation, (6) support for probabilistic risk assessments, (7) design analyses, and (8) nuclear plant training and instrument and control simulators.

With respect to code qualification, the list of targeted applications can be distilled to two key elements: the need to accurately simulate plant type and event type. Thus, with respect to targeted applications, an important source of validation requirements arises from the need to accurately model the response of PWR and BWR plants currently operational in the United States for a spectrum of transient and accident scenarios.

5.1. Plant Type

A survey of commercial nuclear power plants was completed in 1992.[5-1] Similar plants designed by a given vendor were placed in groups characterized by coolant loop configuration, the number of fuel bundles, and bundle design. This information is summarized in Table 5-1 for PWRs; a similar summary is provided in Table 5-2 for BWRs.

5.2. Event Type

It is impossible to list all the potential event scenarios (accidents, transients, and operating events) and correlate these to the accident scenarios simulated in each IET. For our purposes, a more modest goal is set, namely, to create a table of the major PWR and BWR event scenarios for use in selection of IETs. This tabulation is provided in Table 5-3.

5.3. IET Selection Based on Scaling Issues

A significant amount of effort will be required to address the scaling issue. That effort is beyond the scope of the present document. However, a promising approach has been identified as part of the RELAP5 adequacy demonstration for AP600 SBLOCA analyses. Scaling analyses are used to demonstrate the relevancy and sufficiency of the collective experimental database for representing the behavior expected of a given plant design during a selected accident scenario. With this approach, an effort is made to demonstrate that the experimental database is sufficiently diverse that the expected full-plant response is included and that the code calculations are comparable with the corresponding tests in nondimensional space. This demonstration permits conclusions relating to code capabilities, drawn from assessments comparing calculated and measured IET test data, to be extended to the prediction of the full-plant behavior. This is a time- and labor-intensive effort. It appears to be generally applicable, if there are sufficient IET facilities. Some diversity in the scaling approaches used when designing the facilities appears desirable. For the AP600 demonstration just described, there were three such IET facilities.

REFERENCES

5-1. J. C. Determan and C. E. Hendrix, “Survey of Thermal-Hydraulic Models of Commercial Nuclear Power Plants,“ EG&G Idaho, Inc. document EGG-EAST-9031 (December 1992).

TABLE 5-1
SUMMARY OF PWR VENDOR AND REACTOR TYPES

Vendor Group	Group Description	Coolant Loops	Number of Bundles	Bundle Design
Westinghouse				
W1	High-power 4 loop	4	193	17 x 17
W2	Medium-power 4 loop	4	193	17 x 17
W3	Low-power 4 loop	4	193	15 x 15
W4	Unique 4 loop	4	157	15 x 15
W5	Unique 4 loop	4	76	16 x 16
W6	High-power 3 loop	3	157	17 x 17
W7	Medium-power 3 loop	3	157	15 x 15
W8	Low-power 3 loop	3	157	14 x 14
W9	2 loop	2	121	14 x 14
AP600	Advanced passive	2 x 4	145	17 x 17
CE				
C1	Unique	3	217	14 x 14
C2	High power	2 x 4	241	16 x 16
C3	Medium power	2 x 4	217	16 x 16
C4	Unique	2 x 4	217	16 x 16
C5	Low power	2 x 4	217	14 x 14
C6	Unique	2 x 4	204	15 x 15
C7	Unique	2 x 4	177	16 x 16
C8	Unique	2 x 4	133	14 x 14
B&W				
B1	High-power, raised loop	2 x 4	205	17 x 17
B2	Low-power, raised loop	2 x 4	177	15 x 15
B3	Low loop	2 x 4	177	15 x 15

TABLE 5-2
SUMMARY OF BWR REACTOR TYPES

Vendor Group	Group Description	Number of Bundles	Bundle Design
GE/BWR/1	G1	84	11x11
GE/BWR/2	G2	560	8x8
GE/BWR/3	G3, low power	484	8x8
	G4, medium power	580	8x8
	G5, high power	724	8x8; 9x9
GE/BWR/4	G6, low power	368	8x8
	G7, medium power	560; 548	8x8
	G8, high power	764	8x8; 9x9
GE/BWR/5	G9	764	8x8; 9x9
	G10, low power	624	8x8
	G11, medium power	748	8x8
	G12, high power	800	8x8

TABLE 5-3
PWR AND BWR EVENT SCENARIOS SUPPORTING THE SELECTION OF IETS

LWR Type	Scenario
Pressurized water reactor	Large-break LOCA Intermediate-break LOCA Small-break LOCA Steam-generator tube rupture Main-steam-line break Loss-of-offsite power Loss of feedwater Reactor trip Anticipated transient w/o scram Multiple-failure events Accident management scenarios

Boiling water reactor	Large-break LOCA
	Intermediate-break LOCA
	Small-break LOCA
	Transients
	    Pressurization
	    Depressurization
	    Rapid reactivity increase
	    Coolant temperature decrease
	    Instability (power oscillation)
	    Anticipated transient w/o scram

6.0. CODE QUALIFICATION—VALIDATION USING OTHER STANDARD TESTS

As discussed in Section 2.1, this element of validation is conducted by comparing code features and code-calculated results with standards not requiring experimental data. It encompasses tests of code features or functions; comparisons of code-calculated results with equilibrium, concept, and analytical solutions; and tests of the numerical methods used in the code.

The collection of tests selected for this element of the TRAC-M validation test matrix is limited in the sense that it does not now, nor will it ever, constitute a complete test of the TRAC-M code. For example, exact solutions, although setting the highest standard for code validation, exist for only a subset of the physical processes and conditions modeled in TRAC-M. Equilibrium, concept problems, and numerical methods also have limitations, as discussed in subsequent subsections.

The tests selected for the TRAC-M validation test matrix for this element are given in this section. The objective of these tests is to provide increased assurance that TRAC-M code features, algorithms, and equations are correctly programmed. Test problems that focus on specific code features, algorithms, and equations in TRAC-M are either devised or defined. Success metrics are established for each problem, and code output is examined to ensure that the expected results are obtained.

Additional test problems are expected to acquire the status of “other standard tests” as TRAC-M development continues under the multiple-team, multiple-site development format employed by the NRC. These should be added to the validation test matrix in a timely manner.

The categories of problems used in this element are

- features tests,
- equilibrium problems,
- concept problems with known outcomes,
- analytical problems (known solutions), and
- problems to test properties of the numerical solution methods.

Descriptions of each of the categories listed above are given in the following discussions, as well as specific recommendations for tests in each category.

6.1. Features Tests

Three code features have been identified for testing. These features, related to TRAC-M input and output, are

- input file error checking,
- output file (graphics) processing, and
- English units input/output.

The initial set of Features Tests, including development status, is presented in Table 6-1.

TABLE 6-1
FEATURES TEST PROBLEMS

Test	Status
Error checking for input decks	In progress
Graphics process	Input deck(s) to be developed
English units input/output	Input decks exist

6.2. Equilibrium Problems

Equilibrium is a condition of balance among various forces. Several types of equilibrium problems exist. First, there are problems with specified initial and boundary conditions such that all real forcing functions that could drive the system from its specified state are zero-valued. Therefore, as the problem is run, the system should remain in equilibrium, which is the success metric. Second, there are problems in which a small nonequilibrium condition is established and the system returns to equilibrium conditions.

An example equilibrium problem of the first type is a horizontal flow channel containing either single-phase vapor, single-phase liquid, or a mixture of subcooled liquid and a noncondensable gas. The channel is open at each end, and the identical pressure is specified at each end and throughout the channel. All fluid and wall temperatures are specified to be identical. The fluid is static, i.e., zero velocity everywhere and no power generation. A transient is run and the outcome examined. The success metric is that the problem should maintain its initial state (zero velocity and constant, specified temperature) for all timestep sizes and for all time. Deviations from the success metric are to be examined and the causes described.

There are three approaches to creating equilibrium problems that can be used to exercise the code. First, an equilibrium condition can be specified via the problem initial and boundary condition specifications as described in the previous paragraph. Second, small departures from equilibrium can be specified initially, and the problem should approach a known equilibrium state. Adjustment of the gravitational head in a vertical flow channel is an example. Following the initial adjustment, equilibrium is attained. Third, an equilibrium state calculated via a steady-state calculation is rerun as a transient restart using the previously calculated steady-state result.

In general, equilibrium problems test for the absence of coding errors that introduce spurious information into the solution. Ideally, each equilibrium problem is designed to test different features. The cause of the failure is sought if the success metric is not satisfied.

The initial set of Equilibrium Problems, including development status, is presented in Table 6-2.

6.3. Concept Problems

Concept problems are problems for which specific outcomes are known even though the exact solution may not be known. An exact but partial success metric can be defined

TABLE 6-2
EQUILIBRIUM TEST PROBLEMS

ID	Test	Status
O1.1	Horizontal pipe hydro equilibrium	Input decks to be developed
O1.2	Displaced vertical fluid column	Input decks to be developed
O1.3	Static vessel	Input deck exists
O1.4	TRAC-P MS#& Standard Test Matrix Problem[6-1]	Existing decks to be modified
O1.5	TRAC-P Conduction Developmental Assessment Problems[6-1]	Existing decks to be modified
O1.6	Air / water hydro equilibrium	Input decks to be developed
O1.7	Liquid / solute hydro equilibrium	Input decks to be developed
O1.8	Radiative energy exchange	Existing decks to be modified

defined. For example, a symmetric perturbation introduced in a symmetric hardware configuration should be preserved, although the precise propagation and attenuation of the perturbation are not known.Concept problems can be devised for most of the basic-equation models in TRAC-M, including the fluid flow equations (single and two phase), conduction equations, power generation model, control system, and component and special-purpose models. Examples of these problems are

- Simple symmetrical fluid flow situations in pipes and the reactor pressure vessel.
- More complex symmetrical fluid flow situations, such as the primary and secondary sides of a complete PWR at steady-state conditions.
- Symmetrical situations for conduction in solids.
- System descriptions that cause changes in the sign of the fluid speed.
- Restart problems to test that results obtained in an original run are exactly repeated after restart.
- Closed-container problems to test conservation of mass and energy.
- Conduction situations that cause a change in the sign of the heat flux.

All the problems that test fluid flow models and methods will be run with single-phase water, two-phase water, and noncondensable gases. Concept problems will be devised for the equipment-component models.

Concept problems are found in the current TRAC-P Standard Test Matrix.[6-1] One series of problems is an isothermal, abrupt flow-area change, vertical coolant-flow channel. This test series uses six different TRAC-hydraulic-component models, including the 3D vessel model to give the same flow channel geometry. The test is executed with single-phase liquid; single-phase vapor; and a two-phase, liquid-vapor mixture. The combinations of TRAC-hydraulic-component arrangements and fluid states give 18 separate problems. The specific known outcome is that all problems should give the identical result. The magnitude of the specific result may not be known analytically.

The problems already available in the Standard Test Matrix can be augmented by making the flow channels horizontal to eliminate gravity and adding an additional hydraulic node to the center of the flow channels. These modifications would allow additional testing as follows: (1) the horizontal channel models, as noted in Table 6-2 above, would allow equilibrium problems to be run; and (2) symmetric perturbation problems could be tested by initializing the central node at a pressure different from all the other nodes. Additional modifications, such as adding heat conductors and power generation, will expand the range of TRAC-M models and methods tested.

The initial set of Concept Problems, including development status, is presented in Table 6-3.

6.4. Analytical Problems

As used in this document, analytical problems have known, exact solutions. The success metric is both exact and complete in the sense that the precise values of all solution variables are known.

TABLE 6-3
CONCEPT TEST PROBLEMS

ID	Test	Status
O2.1	TRAC-P MS#& Standard Test Matrix Problem[6-1]	Existing decks
O2.2	Symmetric perturbations in the MS#& Standard Test Problems[6-1]	Existing decks to be modified
O2.3	HCOND# Standard Test Matrix Problem[6-1]	Existing decks
O2.4	DRAIN Standard Test Matrix Problem[6-1]	Existing deck
O2.5	ROD2 Standard Test Matrix Problem[6-1]	Existing deck
O2.6	Bubble rise problems	Existing decks
O2.7	Falling drop problems	Existing decks
O2.8	Boron transport problem	Existing decks
O2.9	Restart validation for 1D SET	Existing deck to be modified
O2.10	Restart validation for 3D SET	Input decks to be developed
O2.11	Restart validation for conduction	Input decks to be developed
O2.12	Restart validation for control system	Input decks to be developed
O2.13	Restart validation for equipment component models and methods	Input decks to be developed
O2.14	Restart validation for special purpose models and methods	Input decks to be developed
O2.15	Mass and energy conservation validation	Input decks to be developed

6.4.1. Basic Equation Models

6.4.1.1. Fluid Flow Equations. A number of analytical solutions exist for steady-state, single-phase flows in simple geometries, both with and without heat transfer. Some available analytical solutions include the following.

- Pressure gradient in simple, unheated flow channels (Ref. 6-2, pp. 188-190).
- Temperature gradient in a heated channel (Ref. 6-2, pp. 390-392).
- Flow in variable-area channels such as expanding and contracting nozzles (Ref. 6-2, pp. 485-486).
- Flow in channels with local pressure losses (Ref. 6-2, pp. 219-220).
- Flow in natural-circulation loops such as thermosyphons (Ref. 6-3, pp. 73-76).
- Flows in distribution manifolds (Refs. 6-4 and 6-5).
- Transport of a scalar by a constant-speed flow (Ref. 6-6).
- Transport of a void wave in a two-phase flow with noncondensable gas (Refs. 6-7 and 6-8).
- Transport of a void wave in a two-phase water flow (Refs. 6-7 and 6-8).
- Nusselt condensation on a vertical surface (Ref. 6-2, pp. 415-420).
- Transport of dissolved solids with a liquid (Ref. 6-6).

These problems can be run with subcooled liquid, superheated vapor, and noncondensable gases to check that the special cases are handled correctly. These problems also test the fluid equation of state and other properties of the fluids and the 1D SET numerical solution method. The fluid equation of state is validated in the sense that given the independent variables solved for by the code, a standard tabulation can be used to obtain the reference value for the dependent variables, and these compared with the values from the TRAC equation of state. This validation method can be used also for the fluid transport properties and the properties of the solids.

The information given in the cited references can be used to develop the problem specification. The success metric will be that the TRAC-M calculated results agree with the analytical solution (within prespecified limits) given in the references. Because these are steady-state problems, spatial resolution will be increased to demonstrate that convergence has been attained.

A few transient analytical solutions for the fluid flow equations are available including:

- Startup of the flow of an incompressible fluid in a simple channel (Ref. 6-3, pp. 21-28).
- Draining of liquid from a tank (Ref. 6-2, p. 237).

- The U-tube manometer problem (Ref. 6-2, pp. 229-230).
- The TRAC-P drain and fill test problem (Ref. 6-9).
- Problems that eliminate the momentum balance from consideration.

The last analytical solutions listed refer to the noncondensable gas capabilities in TRAC-M. The perfect gas with variable specific heat modeling for these gases allows derivation of both steady-state and transient analytical solutions. Many of these are given in thermodynamics textbooks. The analytical solution is obtained from the mass and energy equations. Specific examples include closed-container problems that allow testing of conservation of mass and energy and the work term in the energy equations. Other transient analytical solutions may be available in the literature and in reports describing verification and validation problems for other computer software.

As in the case of the steady-state problems, the cited references can be used to develop the problem specification and TRAC-M model. The success metric will be that the TRAC-M calculated results agree with the analytical solution given in the references. User guidance is provided in the form of the requirement to demonstrate temporal and spatial convergence of the TRAC-M numerical solution to the analytical solution.

6.4.1.2. Heat Conduction in Solids. There are numerous analytical solutions available for the heat conduction equation. The TRAC-P Standard Test Matrix report,[6-1] the TRAC Developmental Assessment Manual,[6-9,6-10] and TRAC-P Theory Manual[6-11] all contain a number of conduction equation solutions and comparisons with TRAC-P predictions. Problems for both one-and two-dimensions in both rectangular and cylindrical geometries are used for TRAC-M validation, including the fuel-clad gap model. These and other conduction problems will be used for TRAC-M validation. The test problems now used for TRAC-P assessment will be used for the validation test matrix. Problem specifications such as those in Appendix E will be developed; the success metric is that the TRAC-M calculated results agree with the analytical results. User guidance is the requirement to demonstrate temporal and spatial convergence.

6.4.1.3. Other Basic Equation Models. Analytical solutions for the radiative energy exchange models have been given by Lam[6-12] and these will be part of the TRAC-M validation test matrix. Analytical solution test problems for the 3D vessel model have not yet been devised.

The tabular input for the power generation in the fuel can be validated by outputting the table and comparing the values with the input values. The point-kinetics model and solution method will be validated by comparing TRAC results with results of a calculation with the ORIGEN2[6-13] isotope buildup and depletion computer code.

We are not aware of benchmark problems that isolate a single reactivity feedback mechanism.

6.4.1.4. Properties of Fluids and Solids. The equations used in TRAC to calculate the equation of state (EOS) and other properties of all the fluids and solid materials available in the code can be validated as a part of the analytical solutions as follows. The liquid and vapor EOS properties for water in TRAC, for example, are functions of the independent variables temperature and pressure. The pressure and temperature obtained during a calculation can be used in the equations for water properties used in TRAC to verify that these equations are correctly coded. A standalone version of the TRAC EOS equations can be used for this purpose. Additionally, the EOS properties given by the TRAC equations can be compared with tabulations of standard values to validate the equations used in TRAC. The transport properties for fluids can be verified and validated by use of the same technique.

This same method can be applied to the solid materials as well. The thermal conductivity of a solid uses the temperature as the independent variable, for example. The value of the solid temperature given by TRAC can be used in the equations for thermal conductivity and both results compared with tabulations of standard values.

6.4.2. Equipment Component and Special-Purpose Models

Currently, we don't have specific examples of analytical solutions for all the equipment component and special-purpose models. Additional literature review is needed to locate or help develop analytical solutions. Two analytical solutions for two special-purpose models are given here.

The critical speed for equilibrium single-phase fluid states is known. Problems that reproduce these known critical flow conditions will be executed with the code. The success metric is that the TRAC-M calculated results should agree with the known critical speed. For these steady-state problems, demonstration of spatial convergence provides user guidance.

The generality of the control system elements in TRAC-M allows a variety of situations with analytical solutions to be devised and tested. Simple ordinary differential equations, for example, can be simulated with control system elements. Ordinary differential equations (ODEs) with known analytical solutions have been used to validate some elements of the TRAC-M control system. These same problems will be selected for the validation test matrix. The success metric is that the TRAC-M calculated results must agree with the analytical solution. User guidance is provided by the requirement that convergence to the analytical solution must be demonstrated.

The initial set of Analytical Problems, including development status, is presented in Table 6-4.

6.5. Numerical Methods Test Problems

These tests are used to demonstrate stability and convergence of the numerical methods. Some of the numerical methods tests can be done in conjunction with the analytical solutions discussed in Section 6.4.1 above. The objective is to demonstrate that the numerical solution methods in TRAC-M are stable and will converge to a solution of the basic partial differential equations. The testing provides assurance that the equations are coded correctly and that the numerical method is stable for some conditions. The success metric will be that stability and convergence are demonstrated.

TABLE 6-4
ANALYTICAL AND NUMERICAL METHODS TEST PROBLEMS

ID	Test	Status
O3.1	Pressure gradient in unheated channel	Input decks to be developed
O3.2	Temperature gradient in heated channel	Input decks to be developed
O3.3	Flow in variable-area channel	Input decks to be developed
O3.4	Flow with local pressure loss	Input decks to be developed
O3.5	Flow in natural circulation loops	Input decks to be developed
O3.6	Flow in distribution manifold	Input decks to be developed
O3.7	Transport of a scalar	Input decks to be developed
O3.8	Void “wave” in noncondensable	Input decks to be developed
O3.9	Liquid enthalpy “wave” in two-phase flow	Input decks to be developed
O3.10	Nusselt condensation	Input decks to be developed
O3.11	Solute transport with liquid	Input decks to be developed
O3.12	Incompressible flow startup	Input decks to be developed
O3.13	Tank draining	Input decks to be developed
O3.14	U-tube manometer problem	Existing deck
O3.15	TRAC-P drain and fill problem	Existing deck
O3.16	Transient noncondensable gas problems	Input decks to be developed
O3.17	1D radial conduction solution	Existing decks
O3.18	2D radial plus axial conduction	Existing decks
O3.19	Radiative exchange	Existing decks
O3.20	Equilibrium critical flow	Input decks to be developed
O3.21	Control system solutions	Existing decks
O3.22	Validate Tabular Power Input	Existing decks
O3.23	Validate Point Kinetics Model	Input deck to be developed
O3.24	3D Neutron Kinetics Benchmarks	Input decks to be developed
O3.25	Numerical methods stability and convergence	Input decks to be developed

Convergence is tested by refining the spatial and timestep increments at a fixed ratio, e.g., one-third the Courant limit. Convergence is demonstrated by showing that as the number of spatial nodes increases, the difference between calculated results decreases. A straight flow channel will be used to help focus on the basic aspects of the numerical methods. Both single-phase and two-phase fluid states, with and without wall heat transfer, will be used in the testing.

An example problem is a straight flow channel, initially at constant pressure, and zero fluid speed. At time greater than zero, the pressure at the pipe inlet will be increased. At fixed locations along the channel, the pressure and fluid speed will be plotted as a function of time for each run. To demonstrate convergence, the plots from successive runs should approach a fixed value.

Accuracy of the spatial difference method will be demonstrated by setting up problems in which a scalar is transported by the motion of the fluid. A temperature “wave” will be used for single-phase flow and a void “wave” for two-phase flow. These flows have analytical solutions and have been included under Section 6.4.1 above. The success metric is that the TRAC-M results agree with the analytical results. User guidance is provided by the requirement that convergence be demonstrated.

The initial set of recommended Analytical and Numerical Methods Test Problems, including development status, is presented in Table 6-4.

6.6. Validation Test Matrix—Validation Using Other Standard Tests

The contributions to the TRAC-M validation test matrix by the Other Standard Tests element are summarized in Table 6-5. Generally, the Equilibrium and Concept Problems test that the equations are coded correctly. These tests do not generally point to specific parts of the equations. Successful completion of these tests generally indicates that nothing major is wrong, but the tests do not indicate that everything is right. They are useful as screening indicators that progressing to the next phase of testing is warranted.

The Analytical and Numerical Methods Test Problems test that, for the limited parts of the equations tested, the correct equations are coded. For steady-state, single-phase flow in a pipe, for example, the friction factor must be correct to calculate the analytical solution with the code.

As shown in Table 6-5, the parts of the TRAC-M coded tested by these validation tests consist mainly of the BEM and NSM. All EOS, transport, and thermal-physical properties for all fluids and solids will be validated as a part of these tests. Limited validation of the other models and methods occurs with these tests. As code development continues, tests for other models and methods by equilibrium and concept problems will evolve.

The SET data that will provide validation of some of the flow field models and FFEC, ECM, and SPM in TRAC-M are discussed in the next section of this report.

TABLE 6-5
VALIDATION OF TRAC-M USING OTHER STANDARDS

Category	Subcategory No.	Subcategory Description	Model	Validation by Other Standards Tests: Best	Validation by Other Standards Tests: Candidates
BEM	1	Fluid mass equation	Mass convection	√ **O3.1**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.1-O3.16, O3.20, O3.24
			Mass exchange due to phase change		
	2	Momentum equation	Momentum flux	√ **O3.3**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.1-O3.16, O3.20, O3.24
			Area change	√ **O3.3**	O3.3, O3.4, O3.6, O3.13
			Pressure gradient	√ **O3.1**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.1-O3.16, O3.20, O3.24
			Wall-to-phase momentum exchange	√ **O3.1**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.1-O3.16, O3.20, O3.24
			Interfacial momentum exchange		
			Momentum exchange due to mass exchange		
			Local losses	√ **O3.4**	O1.1, O3.4, O3.6
			Gravity	√ **O3.5**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.5, O3.13, O3.14
	3	Fluid energy equation	Energy convection	√ **O3.2**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.1-O3.16, O3.20, O3.24
			Pressure-work term	√ **O3.16**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.1-O3.16, O3.20, O3.24
			Wall-to-phase energy exchange	√ **O3.2**	O3.2, O3.5, O3.10
			Interfacial energy exchange		
			Direct energy deposition		
			Energy exchange due to mass exchange		
	4	Noncondensable gas and liquid solute	Mass convection	√ **O3.8**	O1.7, O3.8
			Solute mass exchange	√ **O3.11**	O1.6, O2.8, O3.11
	5	3D Vessel model	As in BEM Subcategories 1-4		O1.3, O1.4
	6	Heat conduction equation	Lumped-capacitance model		
			1D radial	√ **O3.17**	O1.5, O2.3, O2.5, O3.17
			2D radial plus axial	√ **O3.18**	O1.5, O2.3, O2.5, O3.18
			Reflood implicit		
			Fuel-clad gap		
			Metal-water reaction		
			Material properties	√ **Any**	O1.5, O2.3, O2.5, O3.17, O3.18

TABLE 6-5 (cont)
VALIDATION OF TRAC-M USING OTHER STANDARDS

Category	Subcategory No.	Subcategory Description	Model	Validation by Other Standards Tests: Best	Validation by Other Standards Tests: Candidates
	7	Power generation in fuel	Tabular power input Point kinetics 3D kinetics Reactivity feedback Fuel temperature Coolant temperature Void fraction Boron concentration	√**O3.22** √**O3.23** √**O3.24**	O3.22 O3.23 O3.24
	8	Radiative energy exchange in the core	Referenced at subcategory level	√ **O3.19**	O1.8, O3.19
	9	Equation of state for fluids	Referenced at subcategory level		All that use fluids
	10	Fluid thermophysical and transport properties	Referenced at subcategory level		All that use fluids
FFEC	1	Regime maps	Bubbly flow Bubbly slug transition Bubbly slug flow Churn flow Annular-mist flow Transition to stratified flow Stratified flow Plug flow		
	2	Fluid mass equation closure (mass exchange)			
	2a	Subcooled boiling	Referenced at subcategory level		
	2b	Interfacial mass exchange	Referenced at subcategory level		
	2c	Plateout of dissolved solids	Referenced at subcategory level		

TABLE 6-5 (cont)
VALIDATION OF TRAC-M USING OTHER STANDARDS

Category	Subcategory No.	Subcategory Description	Model	Validation by Other Standards Tests: Best	Validation by Other Standards Tests: Candidates
FFEC (cont)	3	Fluid momentum equation closure (momentum exchange)			
	3a	Wall-to-phase momentum exchange	Single phase	√ **O3.1**	O1.1-O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O3.1, O3.2, O3.5, O3.6, O3.12
			Two-phase, homogeneous		O2.1
			Two-phase, horizontal stratified		
	3b	Interfacial momentum exchange	Bubbly flow		O2.6
			Bubbly slug transition		
			Bubbly slug flow		
			Churn flow		
			Annular-mist flow		O2.7
			Transition to stratified flow		
			Stratified flow		
			Plug flow		
	3c	Local pressure losses	Abrupt expansion	√ **O3.4**	O1.1, O2.1, O2.2, O3.4
			Abrupt contraction	√ **O3.4**	O1.1, O2.1, O2.2, O3.4
			Orifice plate	√ **O3.4**	O1.1, O3.4
			User supplied	√ **O3.4**	O1.1, O3.4
	4	Fluid energy equation closure (energy exchange)			
	4a	Wall-to-phase energy exchange	Natural convection to liquid		
			Forced convection to liquid	√ **O3.2**	O3.2
			Nucleate boiling		
			Critical heat flux		
			Transition boiling		
			Minimum stable film boiling temperature		
			Film boiling		
			Single-phase vapor		
			Condensation	√ **O3.10**	O3.10
			Two-phase forced convection		

TABLE 6-5 (cont)
VALIDATION OF TRAC-M USING OTHER STANDARDS

Category	Subcategory No.	Subcategory Description	Model	Validation by Other Standards Tests: Best	Validation by Other Standards Tests: Candidates
FFEC (cont)	4b	Interfacial energy exchange	Bubbly flow Bubbly slug transition Bubbly slug flow Churn Flow Annular-mist flow Transition to stratified flow Stratified flow Plug flow Effect of noncondensables		
ECM	1	Centrifugal pumps (Pump component)	Referenced at subcategory level		
	2	Steam-water separator	Referenced at subcategory level		
	3	Plenum component	Referenced at subcategory level		
	4	Valve component	Referenced at subcategory level		
	5	Turbine	Referenced at subcategory level		
	6	Pressurizer	Referenced at subcategory level		
SPM	1	Model for CCFL	Referenced at subcategory level		
	2	Critical flow model	Referenced at subcategory level	√ **O3.20**	O3.20
	3	Trip and control elements	Referenced at subcategory level	√ **O3.21**	O3.21
	4	Reflood heat transfer models			
	4a	Flow regime modeling	Bubbly flow Inverted annular flow Dispersed flow		
	4b	Wall-to-phase fluid drag	Single phase Two-phase homogeneous		

TABLE 6-5 (cont)
VALIDATION OF TRAC-M USING OTHER STANDARDS

Category	Subcategory No.	Subcategory Description	Model	Validation by Other Standards Tests: Best	Validation by Other Standards Tests: Candidates
SPM (cont)	4c	Interfacial fluid drag	Subcooled boiling Smooth inverted annular flow Rough-wavy inverted annular flow Agitated inverted annular flow Post-agitated (dispersed) flow Highly dispersed flow		
	4d	Wall-to-phase fluid heat transfer	Forced convection to a single-phase liquid Nucleate boiling Critical heat flux Transition boiling Min. stable film boiling temperature Film boiling Convection to a single-phase vapor Convection to a two-phase mixture Condensation Natural convection to a single-phase liquid		
	4e	Interfacial fluid heat transfer	Bubbly flow Inverted annular flow Dispersed flow		
	4f	Conduction heat transfer	Referenced at subcategory level		
	5	Two-phase level-tracking model	Referenced at subcategory level		
	6	Offtake model for Tee component	Referenced at subcategory level		
NSM		Fluid field equations	Referenced at subcategory level		
		1D stability enhancing two-step (SETs) method	Referenced at subcategory level	√ **O3.25**	O1.1, O1.2, O1.4, O1.6, O1.7, O2.1, O2.2, O2.4, O2.8, O2.9, O2.15, O3.1-O3.16, O3.20, O3.25
		3D SETs	Referenced at subcategory level		O1.3, O1.4, O2.6, O2.7, O2.10

TABLE 6-5 (cont)
VALIDATION OF TRAC-M USING OTHER STANDARDS

Category	Subcategory No.	Subcategory Description	Model	Validation by Other Standards Tests: Best	Validation by Other Standards Tests: Candidates
NSM (cont)		Conduction in solid materials			
		1D rectangular and cylindrical	Referenced at subcategory level	√ **O3.17**	O1.5, O2.3, O2.5, O2.11, O3.17
		2D rectangular and cylindrical	Referenced at subcategory level	√ **O3.18**	O1.5, O2.3, O2.5, O2.11, O3.18
		Power generation-fuel rods	Tabular power input Point kinetics 3D kinetics Reactivity feedback	- √**O3.22** √**O3.23** √**O3.24**	- O3.22 O3.23 O3.24
		Radiative energy exchange	Referenced at subcategory level	√ **O3.19**	O1.8, O3.19
		Fluid equation of state	Referenced at subcategory level	√	All that use fluids
		Fluid boundary conditions	Referenced at subcategory level	√	All that use fluids
		Equipment component models		-	-
		Pump component	Referenced at subcategory level		
		Steam-water separator	Referenced at subcategory level		
		Plenum component	Referenced at subcategory level		
		Valve component	Referenced at subcategory level		
		Turbine	Referenced at subcategory level		
		Pressurizer	Referenced at subcategory level		
		Special-purpose models			
		Model for CCFL	Referenced at subcategory level		
		Critical flow model	Referenced at subcategory level	√**O3.20**	O3.20
		Trip and control elements	Referenced at subcategory level	√ **O3.21**	O3.21
		Reflood heat transfer	Referenced at subcategory level		
		Steady-state methods	Referenced at subcategory level	√	All Steady-State Problems
		Timestep size and control	Referenced at subcategory level	√	All

REFERENCES

6-1. R. G. Steinke, "A Description of the Test Problems in the TRAC-P Standard Test Matrix," Los Alamos National Laboratory document LA-UR-96-1475 (May 1996).

6-2. R. B. Bird, W. E. Stewart, and E. N. Lightfoot, *Transport Phenomena* (John Wiley & Sons, Inc., New York, New York, 1960).

6-3. N. E. Todreas and M. S. Kazimi, *Nuclear Systems II: Elements of Thermal Hydraulic Design* (Hemisphere Publishing Corporation, New York, New York, 1990).

6-4. R. A. Bajura, "A Model for Flow Distribution in Manifolds," Transactions of the ASME Journal of Engineering for Power (1971), pp. 7–12.

6-5. R. A. Bajura, "Flow Distribution Manifolds," Transactions of the ASME Journal of Fluids Engineering (1976), pp. 7–12.

6-6. J. H. Mahaffy, "Numerics of Codes: Stability, Diffusion, and Convergence," *Nuclear Engineering and Design* **145**, 131–145 (1993).

6-7. M. P. Paulsen, J. G. Shatford, J. L. Westacott, and L. J. Agee, "A Nondiffusive Solution Method for RETRAN-03 Boiling Water-Reactor Stability Analysis," *Nuclear Technology* **100**, No. 2, 162–173 (1992).

6-8. F. J. Moody, *Introduction to Transient Thermofluid Mechanics* (John Wiley & Sons, New York, New York, 1990).

6-9. B. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore and J. L. Steiner, "TRAC-M: Fortran 77, Version 5.5, Developmental Assessment Manual, Volume I: Assessment Sections not including 2D/3D Tests," Los Alamos National Laboratory document LA-UR-99-6480 (December 1999).

6-10. B. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore and J. L. Steiner, "TRAC-M: Fortran 77, Version 5.5, Developmental Assessment Manual, Volume II: Assessment Sections for 2D/3D Tests," Los Alamos National Laboratory document LA-CP-99-345 (December 1999).

6-11. J. W. Spore, S. J. Jolly-Woodruff, T. K. Knight, J-C. Lin, R. A. Nelson, K. O. Pasamehmetoglu, R. G. Steinke, and C. Unal, "TRAC-PF1/MOD2 Theory Manual," Los Alamos National Laboratory document NUREG/CR-5673, Volume 1 (July 21, 1993).

6-12. Kin L. Lam, "Implementation of Radiation Heat-transfer Model into the TRAC-P Code," Los Alamos National Laboratory report LA-UR-96-0174 (October 1995).

6-13. A. G. Croff, "ORIGEN2 - Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code," Oak Ridge National Laboratory report ORNL-5621 (July 1980).

7.0. CODE QUALIFICATION—VALIDATION USING SEPARATE EFFECTS TEST DATA

As discussed in Section 2, this element of validation contributes to code qualification by comparing code-calculated results with SET data. SETs are experiments in which a limited number of physical phenomena of interest occur, and detailed, high-quality data are obtained under closely controlled conditions. SETs cover a spectrum of tests from the most fundamental, to those investigating interactions between phenomena and components or equipment in a specific region of the physical system. The primary use of data from SETs is to assess the adequacy of the closure models and closed form analytical models used in the code.

The summary PIRT (Section 4, Table 4-5 and the other PIRT tables upon which Table 4-5 is based) is the sole source of requirements for the SET element of the TRAC-M validation test matrix.

7.1. SET Element Completion Status

Several features of the TRAC-M validation test matrix reflect work in progress or yet to be accomplished. The objective of this section is to identify the areas of the SET element of the validation test matrix that are incomplete.

With respect to the coverage of PWR LL phenomena, potential validation tests have been identified only for the W-PWR LB LOCA (Table 4-2a). These derive from an earlier LB LOCA validation test matrix effort,[7-1] but do include the highly ranked phenomena from both the AP600 PIRT[7-2] and W four-loop PWR PIRT[7-3] efforts. As seen in the summary tabulation of highly ranked PWR phenomena (Table 4-2d), additional PWR phenomena arise from the other PWR PIRTs, namely the W and B&W SB LOCAs, e.g., transition boiling, condensation on surfaces, and post-CHF heat transfer. SET tests have not yet been identified for these phenomena. In addition, it is anticipated that additional phenomena will be added to the SET validation test matrix as PIRTs are completed for other plants, accidents, and transients.

With respect to the coverage of BWR LL phenomena, potential validation tests have been identified only for the BWR LB LOCA (Table 4-4a). As seen in the summary tabulation of highly ranked BWR phenomena (Table 4-4d), additional BWR phenomena arise from the other BWR PIRTs, namely the SB LOCAs and transient events. SET tests have not yet been identified for these phenomena.

At present, the number of tests entered in the SET validation test matrix may be larger than necessary. This situation exists because data availability is presently uncertain for a number of the tests currently included in the PWR SET element of the TRAC-M validation test matrix. As data availability is determined, it is expected that the SET matrix will be revised accordingly.

7.2. Data Selection Based on PIRT Summary

With a few exceptions, the present TRAC-M analytical and constitutive models used in both PWR and BWR applications derive from the TRAC-P code.[7-4] Work to improve the TRAC-M constitutive models is planned. As this work is completed, the constitutive models will be tested for both PWR and BWR applicability as appropriate.

Referring to the consolidated PIRT (Table 4-5), the LL phenomena can be assigned to one of three groups: highly ranked PIRT phenomena common to both PWRs and BWRs, highly ranked phenomena derived from PWR PIRTs only, and highly ranked phenomena derived from BWR PIRTs only.

The TRAC-M SET validation test matrix is based upon these three groups of PIRT phenomena and consists of three parts. The first part consists of common validation tests that apply to the entirety of the consolidated code, whether used in PWR or BWR application (Section 7.2.1). The second part consists of validation tests that are specific to PWR phenomena (Section 7.2.2). The third part consists of validation tests that are specific to BWR phenomena (Section 7.2.3).

7.2.1. Common SET Validation Tests

Validation tests that apply to the consolidated code, whether used in PWR or BWR applications are listed in Table 7-1. Additional details about the common validation tests included in the SET element of the TRAC-M validation are presented in Appendices F (PWR) and G (BWR), specifically the applicable literature or report citations and the testing ranges for key parameters, if available.

The first column in Table 7-1 identifies the PIRT phenomenon with which the validation tests are associated. The second column is an identifying number for each validation test of the form Sx.y, with the "S" denoting SET, "x" being a number common to all tests for the same PIRT phenomenon, and "y" being the individual identifying number within set "x". The third column identifies the facility, and if applicable, lead investigator. The fourth column contains a brief statement characterizing the key feature of the test. The fifth column contains a symbol to communicate a priority assessment, namely whether the test is deemed vital or desirable. The sixth column provides summary information about the existence of TRAC input models (decks). A "-" is entered if no input model exists. If an input model exists, the deck location, need for updating for use with the current version of the code, and availability of quality assurance documentation are summarized. The seventh column provides summary information about the availability of the test data to be used for the validation exercise. If the availability of the data is unknown, an "-" is entered. If the data are available, additional information about the data is summarized. The eighth and final column cross correlates the facility (column 3) with the corresponding table and reference in Appendix F, e.g., F-12=>1,2 refers to references 1 and 2 in Table F-12, and the identifying number of the facility in the OECD/CSNI separate effects test matrix for thermal-hydraulic code validation,[7-6] should the selected facility be described in that document. A listing of TRAC-M input decks for common SETs is provided in Appendix G.

7.2.2. Additional PWR SET Validation Tests

Additional validation tests that arise from phenomena found to be important only in PWRs are summarized in Table 7-2. The format for Table 7-2 is identical to that of Table 7-1.

Additional details about the additional PWR validation tests included in the SET element of the TRAC-M validation matrix are presented in Appendix F. A listing of TRAC-M input decks for PWR-specific SETs is provided in Appendix G.

7.2.3. Additional BWR SET Validation Tests

Additional validation tests that arise from phenomena found to be important only in BWRs[7-5] are summarized in Table 7-2. The format for Table 7-2 is identical to that of Table 7-1.

Additional details about the additional BWR validation tests included in the SET element of the TRAC-M validation matrix are presented in Appendix H, beginning with Table H-16. In several instances, BWR-specific tests are entered for a phenomenon identified in the common set validation matrix. A listing of TRAC-M input decks for BWR-specific SETs is provided in Appendix I.

REFERENCES

7-1. E. D. Hughes and B. E. Boyack, “TRAC-P Validation Test Matrix,” Los Alamos National Laboratory document LA-UR-97-3990 (September 1997).

7-2. B. E. Boyack, “AP600 LBLOCA Phenomena Identification and Ranking Tabulation,” Los Alamos National Laboratory document LA-UR-95-2718 (1995).

7-3. Technical Program Group, EG&G Idaho, Inc., “Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA,” United States Nuclear Regulatory Commission report NUREG/CR-5249, 1989.

7-4. J. W. Spore, S. J. Jolly-Woodruff, T. K. Knight, J-C. Lin, R. A. Nelson, K. O. Pasamehmetoglu, R. G. Steinke, and C. Unal, “TRAC-PF1/MOD2 Theory Manual,” Los Alamos National Laboratory document NUREG/CR-5673, Volume 1 (July 21, 1993).

7-5. M. Straka and L. W. Ward, “BWR PIRT and Assessment Matrices for BWR LOCA and Non-LOCA Events,” Scientech, Inc. document SCIE-NRC-393-99 (1999).

7-6. Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, Volume I, Phenomena Characterization and Selection of Facilities and Tests; Volume II, Facility and Experiment Characteristics, Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency report NEA/CSNI/R(93)14/Part 1, Part 2/Rev. (September 1993).

TABLE 7-1
COMMON SET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Append. F; OECD/CSNI
Boiling–film	Sc1.1	UoO[a]/Stewart	Fundamental tube data	++	-	4	F-1=>2; -
(Table F-1)	Sc1.2	UoO/Laperriere	Fundamental tube data	+	-	4	F-1=>3; -
	Sc1.3	UoO/Fung	Fundamental tube data	+	-	4	F-1=>5; -
	Sc1.4	Winfrith	Fundamental tube data	++	1,4	2	F1=>4; 10.4
	Sc1.5	THEF[b]/INEL	Fundamental tube data	++	1,4	2	F1=>4; 11.3
	Sc1.6	Lehigh	Fundamental rod-bundle data	++	1,4	2	F1=>4; 11.42
	Sc1.7	TPTF[c]/JAERI[d]	BWR and PWR core geometries	+	-	-	F1=>4; 6.1
	Sc1.8	Blowdown HT/RS37	25-rod bundle	+	-	-	F1=>4; 4.5
Condensation–Interfacial	Sc2.1	Lee	Cocurrent stratified horizontal flow	++	-	4	F-3=>1; -
(Table F-3)	Sc2.2	Kim	Countercurrent steam-water stratified flow	++	-	4	F-3=>2; -
	Sc2.3	Akimoto	Water into flow steam at 90 degree angle	++	1,3,4	4	F-3=>3,4; -
	Sc2.4	Celata	Superheated steam on subcooled water surface	+	-	4	F-3=>5,6; -
Flashing–interfacial	Sc3.1	Critical Flow Facility/GE	Flashing discharge through pipe	++	-	1	F-7=>1; 11.54
	Sc3.2	GE Vessel Test 1004-3	small vessel test w/ void fraction <0.5	++	1	3	H-25=>3; 11.44
	Sc3.3	GE Vessel Test 5801-13	large vessel test	++	1	3	H-25=>3; -
(Table F-7 and H-25)	Sc3.4	Edwards Blowdown	Pipe blowdown	++	1,3,4	2, 4	F-7=>2; 3.15
	Sc3.5	Canon (Initial:Vertical:Super)	Pipe blowdown	++	2,3,4	4	F-7=>3; 3.3, 3.4
	Sc3.6	BNL[e] Nozzle	Converging-diverging nozzle	+	-	1	F-7=>4,5; -
	Sc3.7	Moby Dick, Super Moby Dick	Critical flow in tubes and nozzles	++	2,3,4	4	F-7=>3; 3.1, 3.2
	Sc3.8	OMEGA	Rod bundle blowdown	+	2,3,4	4	F-7=>3; 3.15
Flow–critical	Sc4.1	Super Moby Dick	Vertical upflow, three nozzle configurations	++	2,3,4	4	F-8=>1; 3.2
(Table F-8)	Sc4.2	Rebecca	Vertical downflow, two nozzle configurations	++	-	1	F-8=>1; 3.25
	Sc4.3	Marviken	Multiple nozzle configurations small to big	++	1,4	1,2	F-8=>1; 8.2
	Sc4.4	TPFL[f]/INEL	Tee critical flow	+	-	1	F-8=>1; 11.35

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL[g] or ISL.[h]
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP[i], journal, or conference proceedings.

[a] University of Ontario.
[b] Thermal Hydraulic Experimental Facility.
[c] Two-phase test facility.
[d] Japan Atomic Energy Research Institute.
[e] Brookhaven National Laboratory.
[f] Two-phase flow loop.
[g] Los Alamos National Laboratory.
[h] Information Systems Laboratories.
[i] Code Assessment and Maintenance Program.

TABLE 7-1 (cont)
COMMON SET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Append. F; OECD/CSNI
Flow–critical (Table F-8) (cont)	Sc4.5	Critical Flow Facility/GE	Low quality critical flows using 7 nozzles	++	-	1	F-8=>8; 11.54
	Sc4.6	Edwards Blowdown	Simulates double-ended break of primary pipe	++	1,4	2,3,4	F-8=>9; -
	Sc4.7	Safety Valve/CISEa -SIET	ADS [b] valves tested	++	-	1	F-8=>1; 5.5
	Sc4.8	Valve Blowdown/CEGBc-MEL	Overpressure protection valves for Sizewell B	+	-	-	F-8=>1; 10.21
Heat conductance–fuel-clad gap (Table F-10)	Sc5.1	Modified Pulse Design	Low pressure	++	-	1	F-10=>3,4; -
	Sc5.2	Modified Pulse Design	High pressure	++	-	1	F-10=>5; -
	Sc5.3	Power Burst Facility	Gap conductance Test Series-2	+	-	1	F-10=>6; -
	Sc5.4	Halden Assembly IFA-226	USNRC-OECD Halden Fuel Behavior Test Prog.	+	-	1	F-10=>7,8; -
Heat transfer–forced convect- ion to vapor (Table F-11)	Sc6.1	Babus'Haq	Tests with air rather than steam	++	-	4	F-11=>1; -
	Sc6.2	Davies & Al-Arabi	Tests performed with water	+	-	-	F-11=>2; -
Interfacial shear (Table F-13 and H-11)	Sc7.1	Dadine	Heated tube	++	-	1	F-13=>2; 3.7
	Sc7.2	Pericles	Boil-off in a bundle w/ void fraction <0.9	++	2	3	H-11=>4; 3.8
	Sc7.3	Pericles Cylindrical	Cylindrical 368-rod core	+	-	-	F-13=>2; 3.7
	Sc7.4	Erset Rod Bundle	36-rod bundle	+	-	1	F-13=>2; 3.7
	Sc7.5	Rebecca	Critical Flow	+	-	1	F-13=>2; 3.7
	Sc7.6	TPTF **d**/JAERI	Horizontal two-phase flow and core heat transfer	+	-	4	F-13=>2; 3.7
	Sc7.7	SCTF/JAERI	2D eight fuel-rod bundle	++	1,4	1	F-13=>2; 3.7
	Sc7.8	CCTF/JAERI	3D 32 fuel-rod bundle	++	1,4	1	F-13=>2; 3.7
	Sc7.9	FRIGG/FROJA	Six-rod and 32-rod test sections	+	-	-	F-13=>2; 3.7
	Sc7.10	NEPTUN-1/NEPTUN-2 Reflood	33-rod test section	++	2,3,4	1	F-13=>2; 3.7
	Sc7.11	Achilles Reflood Loop	68-rod test section ballooned and unballooned	+	2,3,4	4	F-13=>2; 3.7
	Sc7.12	THETIS Bundle	7 x 7 rod test section	++	1,2,3,4	4	F-13=>2; 3.7
	Sc7.13	FLECHT-SEASET/W	17 x 17 rod bundle	++	1,4	1	F-13=>2; 3.7

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Centro Informazioni Studi Esperienze, SpA.
[b] Automatic depressurization valve.
[c] Central Electricity Generating Board.
[d] Two-phase test facility.

TABLE 7-1 (cont)
COMMON SET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Append. F; OECD/CSNI
Interfacial shear	Sc7.14	THTF/ORNLe	8x8 rod bundle, steady-state and transient	++	1,3,4	1	F-13=>5; 11.38
(Table F-13 and H-11)	Sc7.15	UPTF/KWU f	1:1 German PWR core simulator	++	1,4	1	F-13=>2; 3.7
(cont)	Sc7.16	1/30;1/15;1/5 Vessel/CREARE	1/15 and 1/30 vessel downcomer tests	++	2,3,4	1	F-13=>2; 3.7
Rewet (Table H-12)	Sc8.1	GOETA Test 42	Test 42; bottom and top reflood	++	2	3	H-11=>2; 8.1
	Sc8.2	NEPTUN	bottom reflood	+	1		H-11=>3; 9.2
	Sc8.3	BWR-FLECHT	Bottom reflood	+	1	3	H-11=>4; 11.23
	Sc8.4	FLECHT-SEASET/W	Bottom reflood	++	1	3	H-11=>5; 11.41

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Oak Ridge National Laboratory.
[b] Kraftwerk Union.

TABLE 7-2
ADDITIONAL PWR SET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Append. F; OECD/CSNI
Boiling–transition (Table F-2)	Sp1.1	UoC [a]/Wang	Fundamental tube and annulus data	++	-	4	F-2=>2,3; -
	Sp1.2	SGTF [b]/ANL	Fundamental tube data	+	-	4	F-2=>4; -
	Sp1.3	UoO/Cheng	Fundamental tube data	+	-	4	F-2=>5; -
	Sp1.4	Johannsen	Fundamental tube data	++	-	4	F-2=>7; -
	Sp1.5	Bennett	Fundamental tube data	++	1,3,4	4	F-2=>8; -
	Sp1.6	FZK [c] Single Rod	Single rod data	+	-	4	F-2=>9,10; -
	Sp1.7	NEPTUN	Rod bundle tests	++	2,3,4	1?	F-2=>9,11/9.2
Draining (Table F-4)	Sp2.1	Foster	Analytical formula for 4 tank geometries	++	-	NA	F-4=>1; -
	Sp2.2	Lubin and Springer	Test drain water from open-top cylinder	++	-	4	F-4=>2; -
	Sp2.3	GIT [d]/Ghiaasiaan	Draining sealed vertical cylinder	++	-	4	F-4=>3,4; -
	Sp2.4	ROSA-AP600	IET experiment	++	2	1	F-4=>5; -
Entrainment/Deentrainment (Table F-5)	Sp3.1	Cousins & Hewitt	Upward flow air-water vertical round tube	+	-	4	F-5=>1,3; -
	Sp3.2	Steen and Wallis	Downward flow air-water in tubes	++	-	4	F-5=>2,3; -
	Sp3.3	Lopez de Bertodano	Adiabatic upward flow air-water loop	++	-	4	F-5=>4,5; -
	Sp3.4	Parabas and Karabelas	Adiabatic horizontal air-water flow	++	-	4	F-5=>6; -
	Sp3.5	Williams	Adiabatic horizontal air-water flow in pipe	+	-	-	F-5=>7; -
Evaporation–interfacial (Table F-6)	Sp4.1	Allesandrini	Steam-water in round vertical tubes	+	-	-	F-6=>2; -
	Sp4.2	Wurtz	Steam-water in tubes and annuli	+	-	-	F-6=>3; -
	Sp4.3	Becker	Single tubes with different heat flux profiles	+	-	-	F-6=>5; -
	Sp4.4	Lehigh	Internal flow in tube using hot patch	++	-	4	F-6=>6,7; 11.57
	Sp4.5	THEF/INEL	Internal flow in heated tube using hot patch	++	1,4	2	F-6=>8,9; 11.3
	Sp4.6	Winfrith	Internal flow in heated tubes	++	1,4	2	F-6=>10-11; -
	Sp4.7	Lehigh	3x3 rod bundle using hot-patch	++	1,4	2	F-6=>12; 11.42

ASSESSMENT NEED:
++ = vital.
\+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] University of Cincinatti.
[b] Steam generator test facility.
[c] Forschungszentrum Karlsruhe.
[d] Georgia Institute of Technology.
[e] Savannah River Laboratory.

TABLE 7-2 (cont)
ADDITIONAL PWR SET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Append. F; OECD/CSNI
Flow–discharge (Table F-9)	Sp5.1	LOFT L3-1	Accumulator discharge	++	1,4	1	F-9=>1; -
	Sp5.2	SRL[e] Gas Pressurizer	Pressurizer discharge	++	-	1	F-9=>2; -
	Sp5.3	KMR-2	Gas-steam pressurizer	+	-	4	F-9=>5; -
Heat transfer–stored energy release (Table F-12)	Sp6.1	Power Burst Facility	Test PCM-2; used unirradiated fuel	+	-	1	F-12=>1,2; -
	Sp6.2	Power Burst Facility	Test LOC-11C	+	-	1	F-12=>3,4; -
	Sp6.3	Phebus LB LOCA	Test 212	+	-	-	F-12=>5; -
	Sp6.4	LOFT	Tests L6-8B01 and L6-8B-2	++	1,4	1	F-12=>6,7;
Noncondensable effects (Table F-15)	Sp7.1	MIT Steam Condensation	Steam condensation with natural circulation	+	-	1	F-15=>1; -
	Sp7.2	MIT[b] Single-Tube Experiment	Steam condensation with forced convection	++	-	1	F-15=>2,3; -
	Sp7.3	UCB Steam Condensation	Steam condensation with natural circulation	++	-	1	F-15=>4,5; -

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Savannah River Laboratory.
[b] Massachusetts Institute of Technology.

TABLE 7-3
ADDITIONAL BWR SET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess Need	TRAC input	Data	Ref: Append. G; OECD/CSNI
Boilingfilm (Table H-1)	Sb1.1	THTF	Test 3.06.6B and Test 3.08.6C	++	1	3	H-1=>1; 11.38
Boiling–nucleate (Table H-2)	Sb2.1	ORNL	Test 3.07.9N	++	-	3	H-2=1; 11.38
Dryout–CHF (Table H-4)	Sb3.1	Biasi		-	-	-	H-4=>1; -
	Sb3.2	CISE		-	-	-	H-4=>2; -
	Sb3.3	Zuber	Apply to countercurrent flow	-	-	-	H-4=>3; -
Flashing–interfacial (Table G5)	Sb4.1	ROSA-III	Tests 901, 902, 924, 926, 905	++	-	3	H-5=>1; -
	Sb4.2	FIST[a]	Test 6DBA1B	++	2	3	H-5=>2; -
Heat–stored (Table H-10)	Sb5.1	See Table 7-1, Common SET Validation Tests: Heat conductance–fuel-clad gap					F-10=>3-8;-
Heat transfer–forced convection to vapor (Table G8)	Sb6.1	THTF bundle	Tests 3.09.10 I, J, K, L. M, N	++	-	3	H-8=>1; 11.38
	Sb6.2	H-2	336 rod bundle uncovery tests 718, 722, 727, 731	-	-	-	H-8=>2; 11.49
Heat transfer–radiation (Table H-9)	Sb7.1	GOETA Test 27	Steady-state experiment in 8x8 bundle	+	2	3	H-9=>1; 8.1
	Sb7.2	THTF	Rod-to-rod and wall during steady state boiloff	++	-	3	H-9=>2; 8.1
Interfacial shear (Table H-11)	Sb8.1	CISE adiabatic pipe	Void fraction>0.5 (CISE-R-291)	++	2	3	H-11=>1; -
	Sb8.2	GE level swell	Tests 1004-3 and 5801-13	++	1	3	H-11=>2; 11.44
	Sb8.3	TLTA[b]-5A	Test 6441	++		3	H-11=>3; -

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Full integral simulation test.
[b] Two-loop test apparatus.

8.0. CODE QUALIFICATION—VALIDATION USING COMPONENT EFFECT TEST DATA

As discussed in Section 2, this element of validation contributes to code qualification by comparing code-calculated results with CET data. Component effect tests investigate behavior in a plant component, frequently but not always at full-scale. Comparisons of code-calculated predictions to data from CETs provide the mechanism for an important aspect of the code qualification effort; these comparisons assess the capability of T-H code to predict component-level phenomena identified in the consolidated PWR and BWR PIRT (Table 4-5). CET data are used to assess the behavior of the integrated code (e.g., field equations, closure relations, component models, numerics, and special models) at the component level.

Component testing can occur in SET, CET or IET facilities.

The summary PIRT (Section 4, Table 4-5 and the other PIRT tables upon which Table 4-5 is based) is the sole source of requirements for the CET element of the TRAC-M validation test matrix.

8.1. CET Element Completion Status

Several features of the TRAC-M validation test matrix reflect work in progress or yet to be accomplished. The objective of this section is to identify the areas of the CET element of the validation test matrix that are incomplete.

With respect to the coverage of PWR CL phenomena, potential validation tests have been identified only for the Westinghouse-PWR LB LOCA (Table 4-2a). These derive from an earlier LB LOCA validation test matrix effort,[8-1] but do include the highly ranked phenomena from both the AP600 PIRT[8-2] and W four-loop PWR PIRT[8-3] efforts. As seen in the summary tabulation of highly ranked PWR phenomena (Table 4-2d), additional PWR CET phenomena arise from the other PWR PIRTs, namely the Westinghouse and B&W SB LOCAs, e.g., flow regime at the break inlet. CET tests have not yet been identified for these phenomena. In addition, it is anticipated that additional phenomena will be added to the CET validation test matrix as PIRTs are completed for other plants, accidents, and transients.

With respect to the coverage of BWR CL phenomena, potential validation tests have been identified only for the BWR LB LOCA (Table 4-4a). As seen in the summary tabulation of highly ranked BWR phenomena (Table 4-4d), additional BWR phenomena arise from the other BWR PIRTs, namely the SB LOCAs and transient events, e.g., multi-channel flows. CET tests have not yet been identified for these phenomena.

At present, the number of tests entered in the CET validation test matrix may be larger than necessary. This situation exists because data availability is presently uncertain for a number of the tests currently included in the TRAC-M validation test matrix. As data availability is determined, it is expected that the SET matrix will be revised accordingly.

8.2. Data Selection Based on PIRT Summary

Several of the TRAC-M plant components, e.g., steam generators and pressurizers, are assembled from more elemental TRAC components. Other components are present in TRAC-M as component models, e.g, pumps, valves and breaks.

Referring to the consolidated PIRT (Table 4-5), the CL phenomena can be assigned to one of three groups: highly ranked PIRT phenomena common to both PWRs and BWRs, highly ranked phenomena derived from PWR PIRTs only, and highly ranked phenomena derived from BWR PIRTs only.

The TRAC-M CET validation test matrix is based upon these three groups of PIRT phenomena and consists of three parts. The first part consists of common validation tests that apply to the entirety of the consolidated code, whether used in PWR or BWR application (Section 8.2.1). The second part consists of validation tests that are specific to PWR phenomena (Section 8.2.2). There are several additional components found in BWRs that are unique to the BWR; they are not present in PWRs. The jet pump is one such component. Also, the BWR fuel assembly configuration differs from that in a PWR; the fuel is contained within a container or can. A separate component model has been incorporated in TRAC-M to model the BWR fuel assembly. The third part consists of validation tests that are specific to BWR phenomena (Section 8.2.3).

8.2.1. Common CET Validation Tests

Validation tests that apply to the consolidated code, whether used in PWR or BWR applications are listed in Table 8-1. Additional details about the common validation tests included in the CET element of the TRAC-M validation are presented in Appendices F (PWR) and G (BWR), specifically the applicable literature or report citations and the testing ranges for key parameters, if available.

The first column in Table 8-1 identifies the PIRT phenomenon with which the validation tests are associated. The second column is an identifying number for each validation test of the form Cx.y, with the “C” denoting CET, “x” being a number common to all tests for the same PIRT phenomenon, and “y” being the individual identifying number within set “x”. The third column identifies the facility, and if applicable, lead investigator. The fourth column contains a brief statement characterizing the key feature of the test. The fifth column contains a symbol to communicate a priority assessment, namely whether the test is deemed vital or desirable. The sixth column provides summary information about the existence of TRAC input models (decks). A “-” is entered if no input model exists. If an input model exists, the deck location, need for updating for use with the current version of the code, and availability of quality assurance documentation are summarized. The seventh column provides summary information about the availability of the test data to be used for the validation exercise. If the availability of the data is unknown, an “-” is entered. If the data are available, additional information about the data is summarized. The eighth and final column cross correlates the facility (column 3) with the corresponding table and reference in Appendix F, e.g., F-16=>1,2 refers to Refs. 1 and 2 in Table F-16, and the identifying number of the facility in the OECD/CSNI separate effects test matrix for thermal-hydraulic code validation,[8-4] should the selected facility be described in that document. A listing of TRAC-M input decks for common CETs is provided in Appendix G.

8.2.2. Additional PWR CET Validation Tests

Additional validation tests that arise from phenomena found to be important only in PWRs are summarized in Table 8-2. The format for Table 8-2 is identical to that of Table 8-1.

Additional details about the additional PWR validation tests included in the CET element of the TRAC-M validation matrix are presented in Appendix F. A listing of TRAC-M input decks for PWR-specific CETs is provided in Appendix G.

8.2.3. Additional BWR CET Validation Tests

Additional validation tests that arise from phenomena found to be important only in BWRs are summarized in Table 8-3. The format for Table 8-3 is identical to that of Table 8-1.

Additional details about the additional BWR validation tests included in the CET element of the TRAC-M validation matrix are presented in Appendix H, beginning with Table H-13. In several instances, BWR-specific tests are entered for a phenomenon identified in the common set validation matrix. A listing of TRAC-M input decks for BWR-specific CETs is provided in Appendix I.

REFERENCES

8-1. E. D. Hughes and B. E. Boyack, "TRAC-P Validation Test Matrix," Los Alamos National Laboratory document LA-UR-97-3990 (September 1997).

8-2. B. E. Boyack, "AP600 LBLOCA Phenomena Identification and Ranking Tabulation," Los Alamos National Laboratory document LA-UR-95-2718 (1995).

8-3. Technical Program Group, EG&G Idaho, Inc., Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA, United States Nuclear Regulatory Commission report NUREG/CR-5249, 1989.

8-4. Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, Volume I, Phenomena Characterization and Selection of Facilities and Tests; Volume II, Facility and Experiment Characteristics, Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency report NEA/CSNI/R(93)14/Part 1, Part 2/Rev. (September 1993).

TABLE 8-1
COMMON CET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Appendix F or G; OECD/CSNI
Flow–countercurrent (Table F-17)	Cc1.1	Dartmouth	Countercurrent flow: steam, subcooled water in vertical tube (fundamental test)	++	1,2,3,4	1,2,3	F-17=>1; 11.16
	Cc1.2	Bankoff	Countercurrent flow: horizontal perforated plate (fundamental test)	++	1,4	1,4	F-17=>5,6
	Cc1.3	1/15; 2/15 BCL[a]	Downcomer countercurrent flow	+	1,3,4	1,2	F-17=>2; 11.4
	Cc1.4	1/30;1/15;1/5 Vessel/CREARE	Downcomer countercurrent flow	++	1,3,4	1	F-17=>3; 11.13
	Cc1.5	1/1; UPTF	Downcomer countercurrent flow; Test 6	++	1	1	F-17=>4; 4.1
	Cc1.6	1/1; UPTF	Upper tie plate countercurrent flow; Test 10C	++	1	1	F-17=>4; 4.1
Flow–multidimensional (Table F-18)	Cc2.1	Rectangular clarifier	Dissertation, University of Windsor	+	-	1	F-18=>4,5; -
	Cc2.2	PERICLES	2D effects in rectangular facility	+	-	-	F-18=>1; 3.8
	Cc2.3	SCTF/JAERI	Runs 718, 719, 720 have multidimensional flow	++	1,2,3,4	1	F-18=>6; 6.14
	Cc2.4	CCTF/JAERI	Run 76 and 76	++	1,2,3,4	1	F-18=>7,8; 6.15
Power–3D distribution Table (H-20)	Cc3.1	ROSA-III	Test 926				H-20=>1; -
Power–decay heat (Table F-20)	Cc4.1	ANS[b] -5.1-1994	American National Standard	++	-	NA	F-5=>1; -
	Cc4.2	AESJ[c]	Proposed Japanese Standard	+	-	NA	F-5=>2; -
	Cc4.3	ISO[d]	Proposed International Standard	+	-	NA	F-5=>3; -
Pressure drop (Table H-22)	Cc5.1	Sher and Greer		++	-	3	H-22=>1; -
	Cc5.2	Muscettola		++	-	-	H-22=>2; -
	Cc5.4	ROSA-III	Test 926	++	-	3	H-22=>4; -
Pump performance (Table F-21)	Cc6.1	SEMISCALE	Radial-flow pump	++	1	1	F-21=>1; 11.39
	Cc6.2	EPRI[e]	Mixed-flow pump	++	-	1	F-21=>2
	Cc6.3	KWU	Axial and mixed-flow pumps; RS 111 project	+	-	4	F-21=>3

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Batelle Columbus Laboratories.
[b] American Nuclear Society.
[c] Atomic Energy Society of Japan.
[d] International Standard Organization.
[e] Electric Power Research Institute.

TABLE 8-2
ADDITIONAL PWR CET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Appendix F or G; OECD/CSNI
Oscillations (Table F-19)	Cp1.1	U-tube manometer	Analytical solution	++	1	2	F-19=>1; -
Reactivity–void (Table F-22)	Cp2.1	None identified					

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

TABLE 8-3
ADDITIONAL BWR CET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Appendix F or G; OECD/CSNI
Flow–channel bypass	Cb1.1	ROSA-III	Tests 901, 926	++	-	3	H-13=>1; -
leakage (Table H-13)	Cb1.2	FIST	Test 6DBA1B	++	-	3	H-13=>2; -
Flow–countercurrent	Cb2.1	BD/ECC[a] /Tobin	Upper tie plate	++	2	3	H-14=>2; -
(Table H-14)	Cb2.2	BD/ECC/Jones	Upper tie plate	++	2	3	H-14=>1,3; -
	Cb2.3	Naitoh	Upper tie plate	+	-	4	H-14=>4; -
	Cb2.4	GOTA	Upper tie plate	+	-	4	H-14=>1,3; -
	Cb2.6	BD/ECC/Jones	Side entry orifice	++	2	1	H-15=>1; -
Flow distribution	Cb3.1	ROSA-III	Tests 901, 902, 926	++	-	3	H-16=>1; -
(Table H-16)	Cb3.2	FIST	Test 6DBA1B	++	-	3	H-16=>2; -
	Cb3.3	TLTA	Tests 6422 (R3); 6423 (R3); 6426 (R1)	++	2	3	H-16=>3; -
	Cb3.4	SSTF[b]	Test EA2-2	++	-	3	H-16=>3; 11.28
Flow–forward	Cb4.1	TLTA-5A	Test 6426/Run 1	++	-	3	H-17=>1; -
(Table H-17)	Cb4.2	FIST	Test 6DBA1B	++	2	3	H-17=>2; -
	Cb4.3	INEL 1/6 jet pump (LSTF[c])	Forward and reverse flow performance	++	1	3	H-17=>3; 11.1
Flow–multidimensional (Table H-18)	Cb5.1	SSTF/UP[d]	Full scale upper plenum; spray into 2-phase mix	++	-	3	H-18=>1; 11.28
Flow–reverse	Cb6.1	TLTA-5A	Test 6426/Run 1	++	-	3	H-19=>2; -
(Table H-19)	Cb6.2	FIST	Test 6DBA1B	++	-	3	H-19=>3; -
	Cb6.3	INEL 1/6 jet pump (LSTF)	Forward and reverse flow performance	++	1	3	H-19=>1; 11.1
Pump performance	Cb9.1	ROSA-III	Test 926	++	-	3	H-23=>1; -
(Table H-23)	Cb9.2	FIST	Test 4DBA1	++	-	3	H-23=>2; -
Spray distrib. (Table H-24)	Cb10.1	SSTF	Full-scale upper plenum	++	-	1	H-24=>1; 11.28

ASSESSMENT NEED:
++ = vital.
\+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Blowdown/emergency core cooling.
[b] Steam sector test facility.
[c] Large-scale test facility.
[d] Upper plenum.

TABLE 8-3 (cont)
ADDITIONAL BWR CET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Appendix F or G; OECD/CSNI
Void distribution (Table H-25)	Cb11.1	Frigg	Boiling in 6x6 bundle	+	-	-	H-25=>1; 8.3
	Cb11.2	GE level swell	Test 1004-3, test 5801-13	++	2	3	H-25=>3; 11.44
	Cb11.3	SSTF/LP [a]	Mixing in lower plenum	++	-	1	H-25=>4; -
	Cb11.4	TLTA	Test 6424/Run1	++	-	3	H-25=>6; -
	Cb11.5	FIST	Test 4DBA1	++	-	3	H-25=>7; -
	Cb11.6	ANL [b] /Marchaterre	Subcooled and saturated void (ANL-5735)	++	1	3	H-25=>5 -

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck QA documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Lower plenum.
[b] Argonne National Laboratory.

9.0. CODE QUALIFICATION—VALIDATION USING INTEGRAL EFFECT TEST DATA

As discussed in Section 2, IETs investigate behavior in a full nuclear power plant, often in a reduced-scale experimental test facility. Comparisons of code-calculated predictions to data from IETs provide the mechanism for three important code qualification efforts. First, IET data are selected to assess the capability of T-H codes to predict SL phenomena identified in the consolidated PIRT (Section 4, Table 4-5). In this manner, IET data are used to determine whether the behavior of the integrated code (e.g., field equations, closure relations, component models, numerics, and special models) adequately simulates highly ranked SL phenomena. Second, IET data are selected to ensure that the code targeted applications are represented (i.e., plant types and accident scenarios). Simulation requirements for plant and targeted application simulation requirements are presented in Section 5. Third, IET data are selected to address scaling issues. If possible, the selected IET facilities should cover a sufficiently broad spectrum of facility scales and transient types to support arguments of code applicability for full-size plants.

9.1. IET Element Completion Status

Several features of the TRAC-M validation test matrix reflect work in progress or yet to be accomplished. The objective of this section is to identify the areas of the IET element of the validation test matrix that are incomplete.

Identification of individual IETs for the TRAC-M validation test matrix arises from the fulfillment of two requirements. The first requirement is that the code be validated by comparison to SL data for highly ranked SL phenomena. The second requirement is that code adequacy be demonstrated for a representative collection of plant types and applications. The relationship between SL PIRT and plant type and targeted applications was illustrated in Fig. 1-2.

With respect to the coverage of PWR SL PIRT phenomena, potential validation tests have been identified for the Westinghouse-PWR LB LOCA (Table 4-2a). These derive from an earlier LB LOCA validation test matrix effort[9-1] but include the highly ranked phenomena from both the AP600 PIRT[9-2] and W four-loop PWR PIRT[9-3] efforts. As seen in the summary tabulation of highly ranked PWR phenomena (Table 4-2d), an additional PWR IET. It is anticipated that additional phenomena will be added to the IET validation test matrix as PIRTs are completed for other plants, accidents, and transients.

With respect to the coverage of BWR SL PIRT phenomena, potential validation tests have been identified only for the BWR LB LOCA (Table 4-4a). As seen in the summary tabulation of highly ranked BWR phenomena (Table 4-4d), additional BWR phenomena arise from the other BWR PIRTs, e.g., carry under flow, pressure wave propagation, and thermal-hydraulic stability. IET tests have not yet been identified for these phenomena.

At present, the number of tests entered in the IET validation test matrix via PIRT SL requirements may be larger than necessary. This situation exists because data availability is presently uncertain for a number of the tests currently included in the

TRAC-M validation test matrix. As data availability is determined, it is expected that the IET matrix will be revised accordingly.

The coverage of PWR and BWR plants and targeted applications in the IET portion of the TRAC-M validation test matrix is believed to be adequate.

9.2. Data Selection Based on PIRT Summary

Referring to the consolidated PIRT (Table 4-5), the SL phenomena can be assigned to one of three groups: highly ranked PIRT phenomena common to both PWRs and BWRs, highly ranked phenomena derived from PWR PIRTs only, and highly ranked phenomena derived from BWR PIRTs only.

The TRAC-M IET validation test matrix is based on these three groups of PIRT phenomena and consists of three parts, one of which contains no IETs at the present time. The first part consists of common validation tests that apply to the entirety of the consolidated code, whether used in PWR or BWR application (Section 9.2.1). The second part consists of validation tests that are specific to PWR phenomena (Section 9.2.2). The third part, if following the pattern of the SET and CET matrices, would consist of validation tests that are specific to BWR phenomena. However, all BWR specific IET phenomena in Table 4-5 arise from PIRTs other than a BWR LB LOCA. As discussed in the previous section, potential validation tests have been identified only for the BWR LB LOCA and thus there are no PIRT required BWR specific IET in this release of the TRAC-M validation test matrix.

Validation tests that apply to the consolidated code, whether used in PWR or BWR applications are listed in Table 9-1. Additional details about the common validation tests included in the IET element of the TRAC-M validation are presented in Appendices F (PWR) and H (BWR), specifically the applicable literature or report citations and the testing ranges for key parameters, if available. A listing of TRAC-M input decks for common and PWR-specific IETs is provided in Appendix G. A listing of TRAC-M input decks for BWR-specific IETs is provided in Appendix I.

The first column of Table 9-1 identifies the PIRT-related or application-related test type. The second column is an identifying number for each validation test of the form Ix.y, with the “I” denoting IET, “x” being a number common to all tests for the same PIRT phenomenon, and “y” being the individual identifying number within set “x”. The third column identifies the facility, and if applicable, lead investigator. The fourth column contains a brief statement characterizing the key feature of the test. The fifth column contains a symbol to communicate a priority assessment, namely whether the test is deemed vital or desirable. The sixth column provides summary information about the existence of TRAC input models (decks). A “-” is entered if no input model exists. If an input model exists, the deck location, need for updating for use with the current version of the code, and availability of quality assurance documentation are summarized. The seventh column provides summary information about the availability of the test data to be used for the validation exercise. If the availability of the data is unknown, an “-” is entered. If the data is availability, additional information about the data is summarized. The eighth and final column cross correlates the facility (column 3) with the identifying number of the facility in the OECD/CSNI separate effects test matrix for thermal-hydraulic code validation.[9-4]

PWR IET validation tests that apply to the consolidated code are listed in Table 9-2.

No BWR IET validation tests that apply to the consolidated code are presently identified as discussed above.

9.3. Data Selection Based on Plant Type and Targeted Applications

T-H codes are specifically designed for a variety of targeted applications. Among these applications are (1) reactor safety analyses for both operating and planned reactors, (2) audits of licensee's calculations, (3) analyses of operating reactor events, (4) analyses of accident management strategies, (5) support for test planning and interpretation, (6) support for probabilistic risk assessments, (7) design analyses, and (8) nuclear plant training and instrument and control simulators.

With respect to code qualification, the list of targeted applications can be distilled to two key elements: plant type and event type.

9.3.1. Plant Type

A survey of commercial nuclear power plants was completed in 1992.[9-5] Similar plants designed by a given vendor were placed in groups characterized by coolant loop configuration (PWR only), the number of fuel bundles, and bundle design. This information is summarized in Table 9-3.

IET facilities based upon W plants have been designed and operated, e.g., Semiscale, LOFT, LSTF, LSTF-AP600, SPES, SPES-AP600, SCTF, CCTF, and UPTF. IET facilities based upon B&W plants have been designed and operated, e.g., MIST, UMCP, and once-through integral system (OTIS) have been designed and operated. The authors are unaware of any IET facilities for CE designs. The use of the W IET facility matrix as a surrogate for the CE plants may be possible.

A listing of TRAC-M input decks for PWR plants is provided in Appendix G.

IET facilities based upon GE-designed BWR plants have been designed and operated, e.g., FIST and ROSA-III. Reasonable coverage of each of the PWR and BWR designs is possible, although each facility has some atypicalities relative to the reference reactor type for which they were designed.

A listing of TRAC-M input decks for BWR plants is provided in Appendix I.

9.3.2. Event Type

It is impossible to list all the potential event scenarios (accidents, transients, and operating events) and correlate these to the accident scenarios simulated in each IET. For our purposes, a more modest goal is set, namely, to create a table of the major event scenarios and an applicable IET facility and a test to represent each scenario. This tabulation is provided for the W and B&W designs in Table 9-4.

With the exception of the SGTR and MSLB transients, TRAC-M PWR performance can be tested for the listed event scenarios for W plants using existing TRAC-P input decks.* Coverage can be provided for these two remaining transients by preparing BETHSY (SGTR) and LOBI (MSLB) facility models, but a cost-benefit assessment should be made, unless TRAC-M input models are required for these facilities for other reasons. With the exception of the LB LOCA, MSLB, loss-of-feedwater event, and ATWS, TRAC-M performance can be assessed for the listed event scenarios for B&W plants.

The companion BWR event scenarios (accidents, transients, and operating events) for which validation tests have been identified are presented in Table 9-5.

9.4. IET Selection Based on Scaling Issues

A significant amount of effort will be required to address the scaling issue. That effort is beyond the scope of the present document. However, a promising approach has been identified as part of the RELAP5 adequacy demonstration for AP600 SB LOCA analyses.[9-6] Scaling analyses are used to demonstrate the relevancy and sufficiency of the collective experimental database for representing the behavior expected of a given plant design during a selected accident scenario. With this approach, an effort is made to demonstrate that the experimental database is sufficiently diverse that the expected full-plant response is included and that the code calculations are comparable with the corresponding tests in nondimensional space. This demonstration permits conclusions relating to code capabilities, drawn from assessments comparing calculated and measured IET test data, to be extended to the prediction of the full-plant behavior. This approach appears to be generally applicable, if there are sufficient IET facilities. For the AP600 demonstration just described, there were three such IET facilities.

REFERENCES

9-1. E. D. Hughes and B. E. Boyack, “TRAC-P Validation Test Matrix,” Los Alamos National Laboratory document LA-UR-97-3990 (September 1997).

9-2. B. E. Boyack, “AP600 LBLOCA Phenomena Identification and Ranking Tabulation,” Los Alamos National Laboratory document LA-UR-95-2718 (1995).

9-3. Technical Program Group, EG&G Idaho, Inc., “Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA,” United States Nuclear Regulatory Commission report NUREG/CR-5249 (1989).

9-4. “Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, Volume I, Phenomena Characterization and Selection of Facilities and Tests; Volume II, Facility and Experiment Characteristics,” Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency report NEA/CSNI/R(93)14/Part 1, Part 2/Rev. (September 1993).

* With few exceptions, existing TRAC-P input decks will require modification for the specific test, even though a TRAC-P input deck exists for the facility.

9-5. J. C. Determan and C. E. Hendrix, "Survey of Thermal-Hydraulic Models of Commercial Nuclear Power Plants," EG&G Idaho, Inc. document EGG-EAST-9031 (December 1992).

9-6. C. D. Fletcher, P. D. Bayless, C. B. Davis, M. G. Ortiz, T. K. Larson, S. M. Sloan, R. A. Shaw, R. R. Schultz, C. E. Slater, G. W. Johnsen, L. S. Ghan, and D. E. Bessette, "Adequacy Evaluation of RELAP5/MOD3, Version 3.2.1.2 for Simulating AP600 Small Break Loss-of-Coolant Accidents" (final draft), Idaho National Engineering Laboratory document INEL-96/0400 (December 1996).

9-7. C. L. Nalezny, "Summary of Nuclear Regulatory Commission's LOFT Program Experiments," Idaho National Engineering Laboratory document NUREG/CR-3214, also EGG-2248 (July 1983).

9-8. "CSNI Integral Test Facility Validation Test Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients," Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency report CSNI 132, Revision 6, restricted working draft (June 1995).

9-9. B. Faydide et al., "Multiple Steam Generator Tube Rupture on the BETHSY Integral Test Facility," ANS International Topical Meeting on Safety of Thermal Reactors, Portland, Oregon (July 21-25, 1999).

9-10. H. Kumararu, I. Ohtsu, Y. Kukita, and L. S. Ghan, "Quick Look Report for ROSA/AP600 Experiment AP-SL-01," Japan Atomic Energy Research Institute document JAERI-memo 07-244 (November 1995).

9-11. J. C. Barbier et al., "A Single Steam Generator Tube Rupture with Unavailability of both the High Pressure Safety Injection System and the Steam Generator Auxiliary Feedwater System on BETHSY Integral Test Facility," International Conference on New Trends in Nuclear System Thermohydraulics, Pisa, Italy (May 30-June 2, 1994).

9-12. P. S. Damerell and J. W. Simons (editors), "2D/3D Program Work Summary Report," MPR Associates document NUREG/IA-0126 (June 1993).

9-13. J. R. Gloudemans, "Multiloop Integral System Test (MIST): Final Report, Volumes 1–10," Babcock & Wilcox Nuclear Power Division document NUREG/CR-5395 (April 1991).

9-14. M. Straka and L. W. Ward, "BWR PIRT and Assessment Matrices for BWR LOCA and Non-LOCA Events," Scientech, Inc., document SCIE-NRC-393-99 (1999).

9-15. "Test 6DBA1B: BWR FIST: Phase 1 Results," United States Nuclear Regulatory Commission report NUREG/CR-3711 (March 1985).

9-16. K. Tasaka et. al., "ROSA-III Double-Ended Break Test Series for a Loss-of-Coolant Accident in a BWR," *Nuclear Technology*, **68**, January 1985, pp. 77-93.

9-17. "BWR BD/ECC Program," United States Nuclear Regulatory Commission report NUREG/CR-2229.

9-18. "BWR FIST: Phase 2 Results," United States Nuclear Regulatory Commission report NUREG/CR-4128 (March 1986).

9-19. "BWR Refill-Reflood Program Task 4.4," United States Nuclear Regulatory Commission report NUREG/CR-2786 (May 1983).

9-20. "SOAR: State of the Art Report on BWR Stability," OECD/CSNI (September 1996).

TABLE 9-1
COMMON IET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess. Need	TRAC Input	Data	Ref: Appendix F or G; OECD/CSNI
Flow–natural circulation (Table H-26)	Ic1.1	ROSA-III/JAERI	Test NC-1 through NC-5	++	-	3	H-26=>2,3; -
	Ic1.2	FRIGG	Tests FT 36a-c	+	-	1, 4	H-26=>1; 8.3
	Ic1.3	FIST	Test 6PNCI-4	+	2	3	H-26=>4-6; -
Level (Table F-14 or G-19)	Ic2.1	Vertical Canon	Vertical tube during blowdown	+	1	1,4	F-14=>2; 3.4
	Ic2.2	Tapioca	Vertical tube–top, middle, and bottom breaks	+	-	1,4	F-14=>2; 3.6
	Ic2.3	Single Tube Level Swell	Vertical heated tube steady-state level swell tests	+	-	-	F-14=>2; 10.14
	Ic2.4	Shoukri Subcooled Boiling	Vertical annular channel	+	-	4	F-14=>4; -
	Ic2.5	Marviken	Test T-11is a level swell experiment	+	1	1	F-14=>2; 8.2
	Ic2.6	GE Level Swell	Tests 1004-3, 5801-13	++	1	3	F-14=>2; 11.44
	Ic2.7	TPTF/ROSA IV/JAERI	Core heat transfer, BWR and PWR cores	+	-	4	F-14=>2; 6.1
	Ic2.8	Creare	1/15 and 1/30 scale vessel downcomer tests	+	1	1	F-14=>2; 6.15
	Ic2.9	UPTF	1:1 German PWR core simulator	++	1	1,2	F-14=>2; 4.1
	Ic2.10	Thetis	7 x 7 test section including level swell tests	+	1	1	F-14=>2; 10.2
	Ic2.11	CCTF/JAERI	Full height 3-D 32-fuel-rod bundle core	++	1	1,2	F-14=>2; 6.15
	Ic2.12	ECN[a] Reflood and Boildown	36-rod test section, boiloff and reflood tests	+	-	-	F-14=>2; 7.1, 7.2
	Ic2.13	FRIGG	36-rod test section	+	-	1, 4	F-14=>2; 8.3
	Ic2.14	NEPTUN-1 Boiloff	33-rod test section, boil-off and reflood tests	+	2	1, 4	F-14=>2; 9.1
	Ic2.15	Pericles Cylindrical	Cylindrical 368-rod core	+	-	-	F-14=>2; 3.9
	Ic2.16	Achilles Reflood Loop	ISP-25	+	2	1, 4	F-14=>2; 10.1
	Ic2.17	FIST	Test 6DBA1B-large recirculation line break	++	2	3	G-19=>2; -

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Energieonderzoek Centrum Nederland.

TABLE 9-2
ADDITIONAL PWR IET VALIDATION TESTS

PIRT Phenomenon (Appendix Table)	No.	Facility/Originator	Test Feature	Assess Need	TRAC Input	Data	Ref: Appendix F or G; OECD/CSNI
Asymmetries (Table F-16)	Ip1.1	LOFT	Test L2-5	++	1,4	1	F-16=>1,2
Oscillations (Table F-19)	Ip2.1	FRIGG Dynamic Tests	Tests 662101, 662105, 662107, 662113, 462053, 462101	++	-	4	F-19=>2-4; 8.3
	Ip2.2	FLECHT-SEASET/W	Test 33437	+	1,4	1	F-19=>5-7; 11.23
	Ip2.3	SCTF/JAERI	Test S2-08	+	1,4	1	F-19=>8,9; 6.15

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

TABLE 9-3
SUMMARY OF VENDOR AND REACTOR TYPES

Type Vendor Group	Group Description	Number of Plants	Coolant Loops	Number of Bundles	Bundle Design
PWR					
Westinghouse					
W1	High-power 4-loop	2	4	193	17 x 17
W2	Medium-power 4-loop	26	4	193	17 x 17
W3	Low-power 4-loop	5	4	193	15 x 15
W4	Unique 4-loop	1	4	157	15 x 15
W5	Unique 4-loop	1	4	76	16 x 16
W6	High-power 3-loop	8	3	157	17 x 17
W7	Medium-power 3-loop	5	3	157	15 x 15
W8	Low-power 3-loop	1	3	157	14 x 14
W9	2 loop	5	2	121	14 x 14
AP600	Advanced passive	0	2 x 4	145	17 x 17
CE					
C1	Unique	1	3	217	14 x 14
C2	High-power	4	2 x 4	241	16 x 16
C3	Medium-power	3	2 x 4	217	16 x 16
C4	Unique	1	2 x 4	217	16 x 16
C5	Low-power	4	2 x 4	217	14 x 14
C6	Unique	1	2 x 4	204	15 x 15
C7	Unique	1	2 x 4	177	16 x 16
C8	Unique	1	2 x 4	133	14 x 14
B&W					
B1	High-power, raised-loop	3	2 x 4	205	17 x 17
B2	Low-power, raised-loop	1	2 x 4	177	15 x 15
B3	Low-loop	7	2 x 4	177	15 x 15
BWR					
GE					
G1	BWR/1	1	NA	84	11 x 11
G2	BWR/2	2	NA	560, 532	8 x 8
G3	Low-power BWR/3	3	NA	484	8 x 8
G4	Medium-power BWR/3	2	NA	580	8 x 8
G5	High-power BWR/3	4	NA	724	8 x 8, 9 x 9
G6	Low-power BWR/4	2	NA	368	8 x 8
G7	Medium-power BWR/4	5	NA	560, 548	8 x 8
G8	High-power BWR/4	11	NA	764	8 x 8, 9 x 9
G9	BWR/5	4	NA	764	8 x 8, 9 x 9
G10	Low-power BWR/6	2	NA	624	8 x 8
G11	Medium-power BWR/6	2	NA	748	8 x 8
G12	High-power BWR/6	1	NA	800	

TABLE 9-4
IET VALIDATION TESTS FOR PWR PLANTS AND TARGETED APPLICATIONS

Plant Type	No.	Event	IET Facility and Test	Assess Need	TRAC Input	Data	Reference: OECD/CSNI
Westinghouse	Pw1.1	LB LOCA	LOFT L2-3 or L2-5	++	1, 3	1	9-7; 9-8
	Pw1.2	IB[a] LOCA	LOFT L5-1 or L8-2	+	1, 3	1	9-7; 9-8
	Pw1.3	SB LOCA	LOFT L3-5 or L3-6	++	1, 3	1	9-7; 9-8
	Pw1.4	SGTR	BETHSY[b] 4.3b	++	-	4	9-9; 9-8
	Pw1.5	MSLB	LOBI BT12	++	2	-	- ; 9-8
	Pw1.6	LOSP[c]	LOFT L9-4	++	1, 3	1	9-7; 9-8
	Pw1.7	Loss of feedwater	LOFT L9-1/L3-3	++	1, 3	1	9-7; 9-8
	Pw1.8	Reactor trip	LOFT L6-2	+	1, 3	1	9-7; 9-8
	Pw1.9	ATWS	LOFT L9-3 or L9-4	+	1, 3	1	9-7; 9-8
	Pw1.10	Multiple failure events	LSTF-AP600 AP-SL-01	+	1, 3	1	9-10; -
	Pw1.11	Accident management	BETHSY 9.3	+	-	4	9-11 ; 9-8
Babcock & Wilcox	Pb1.1	LB LOCA	CCTF C2-10 (vent-valve test)	++	1,3	1	9-12; 9-8
	Pb1.2	IB LOCA	MIST 4100B2	++	1	1	9-13; 9-8
	Pb1.3	SB LOCA	MIST 3109AA	++	1	1	9-13; 9-8
	Pb1.4	SGTR	MIST 3404AA	++	1	1	9-13; 9-8
	Pb1.5	MSLB	None available	-	-	-	-
	Pb1.6	LOSP	MIST 4SB011	++	1	1	9-13; 9-8
	Pb1.7	Loss of feedwater	None available	-	-	-	-
	Pb1.8	Reactor trip	MIST 4SB011	++	1	1	9-13; 9-8
	Pb1.9	ATWS	None available	-	-	-	-
	Pb1.10	Multiple failure events	MIST 410BD1 or 410 AT3	+	1	1	9-13; 9-8
	Pb1.11	Accident management	MIST 410BD1 or 410AT3	+	1	1	9-13; 9-8

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Intermediate break.
[b] Boucle d'Etudes Thermohydrauliques Système.
[c] Loss of offsite power.

TABLE 9-5
IET VALIDATION TESTS FOR BWR PLANTS AND TARGETED APPLICATIONS[9-14]

Plant Type	No.	Event	IET Facility and Test or Plant	Assess Need	TRAC Input	Data	Reference; OECD/CSNI
		LOCA	**IET FACILITY**				
BWR/6	Pb1.1	Large recirculation line	FIST 6DBA1B	++	1	3	9-15; 9-8
BWR/4	Pb1.2	Large recirculation line	FIST 4DBA1	++		3	9-14 ; 9-8
BWR/general	Pb1.3	Large recirculation line	FIX-II Test 3061				9-14 ; 9-8
BWR/general	Pb1.4	Large recirculation line	ROSA-III Run 901	++		3	9-16; 9-8
BWR/general	Pb1.5	Large recirculation line	ROSA-III Run 905			3	9-16; 9-8
BWR/general	Pb1.6	Large recirculation line	ROSA-III Run 902	+		3	9-16; 9-8
BWR/general	Pb1.7	Large recirculation line	ROSA-III Run 924			3	9-16; 9-8
BWR/general	Pb1.8	Large recirculation line	ROSA-III Run 926			3	9-14 ; 9-8
BWR/general	Pb1.9	Large recirculation line	TBL [a] Test 108				9-14 ; 9-8
BWR/general	Pb1.10	Large recirculation line	TLTA 6422 Run 3			3	9-17; 9-8
BWR/general	Pb1.11	Large recirculation line	TLTA 6424 Run 1			3	9-17; 9-8
BWR/general	Pb1.12	Large recirculation line	TLTA 6423 Run 3	++	1	3	9-17; 9-8
BWR/general	Pb1.13	Large recirculation line	TLTA 6426 Run 1	+		3	9-17; 9-8
BWR/6	Pb1.14	Medium recirculation line	FIST 6IB1			3	9-14 ; 9-8
BWR/6	Pb1.15	Medium recirculation line	FIST 6LB1A			3	9-18; 9-8
BWR/general	Pb1.16	Medium recirculation line	ROSA-III Run 962				9-16; 9-8
BWR/general	Pb1.17	Refill/reflood	Piper-ONE PO-LB-1				9-14 ; 9-8
BWR/general	Pb1.18	Refill/reflood	SSTF	++		3	9-19; 9-8
BWR/6	Pb1.19	Small recirculation line	FIST 6SB1	++	2	3	9-14 ; 9-8
BWR/general	Pb1.20	Small recirculation line	Piper-ONE PO-SB-7				9-14 ; 9-8
BWR/general	Pb1.21	Small recirculation line	ROSA-III Run 912	++		3	9-14 ; 9-8
BWR/general	Pb1.22	Small recirculation line	ROSA-III Run 984	+		3	9-14 ; 9-8
BWR/general	Pb1.23	Small recirculation line	TBL Test 311				9-14 ; 9-8
BWR/general	Pb1.24	Small recirculation line	TLTA 6432 Run 1			3	9-14 ; 9-8
BWR/6	Pb1.25	Steam line break	FIST 6MSB1	++		3	9-14 ; 9-8
BWR/general	Pb1.26	Steam line break	ROSA-III Run 953	+		3	9-14 ; 9-8
BWR/general	Pb1.27	Steam line break	TBL Test 314				9-14 ; 9-8

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Two-bundle loop.

TABLE 9-5 (cont)
IET VALIDATION TESTS FOR BWR PLANTS AND TARGETED APPLICATIONS[9-14]

Plant Type	No.	Event	IET Facility and Test or Plant	Assess. Need	TRAC Input	Data	Reference; OECD/CSNI
		TRANSIENT					
BWR/6	I28	ATWS MSIV[a] closure	FIST 6PMC2A	++	1	3	9-15; 9-8
	I29	Water level drop	FIST T23C	+		3	9-14 ; 9-8
BWR/6	I30	Controlled depress.	FIST 6PMC3	++		3	9-14 ; 9-8
BWR/6	I31	Natural circulation	FIST 6PNC1	+	2	3	9-15; 9-8
BWR/6	I32	Natural circulation	FIST 6PNC3			3	9-18; 9-8
	I33	Natural circulation	ROSA-III NC-1 …NC-5	++		3	9-16; 9-8
	I34	Water level drop	FIST T1QUV			3	9-14 ; 9-8
BWR/4	I35	Turbine trip	FIST 4PTT1	++	2	3	9-14 ; 9-8
			PLANT				
BWR/4	P1	AOT: feedwater trip	Browns Ferry	+	1	3	
BWR/4	P2	Load rejection	Browns Ferry	+	1	3	
BWR/4	P3	Reactor coolant pump trip	Browns Ferry	+	1	3	
BWR/GETSCO reactor	P4	MSIV closure	Leibstadt	+	2		9-14 ; 9-8
BWR/GETSCO reactor	P5	Feedwater loss	Leibstadt	+	2		9-14 ; 9-8
BWR/4	P6	turbine trip	Peach Bottom-2	++	2	3	9-14 ; 9-8
		STABILITY					
	P7		Dodeward		1	3	9-20; -
BWR/5	P8		LaSalle-2	++	1		9-20; -
BWR/GETSCO reactor	P9		Leibstadt	+	2		9-20; -
BWR/ABB Atom reactor	P10		Ringhals-1	++	2		9-20; -
BWR/5	P11		WNP-2[b]	+			9-20; -

ASSESSMENT NEED:
++ = vital.
+ = desirable.

TRAC INPUT:
1 = exists/available at LANL or ISL.
2 = exists/outside LANL or ISL.
3 = deck will require updating.
4 = deck quality assurance documentation unavailable.

DATA:
1 = available NUREG/CR, NUREG, NRC or OECD/CSNI databank, or equiv.
2 = available at LANL.
3 = available at ISL.
4 = limited data: NUREG/IA, CAMP, journal, or conference proceedings.

[a] Main steam isolation valve.
[b] Washington Nuclear Power Unit 2.

APPENDIX A

VALIDATION SUCCESS METRICS

Validation is defined in this report as the comparison of code predictions to standards, either experimental data or other. The success metrics are the same as those used in the recently completed RELAP5 adequacy assessment effort;[A-1] they are repeated here for convenience.

"Excellent agreement" applies when the code exhibits no deficiencies in modeling a given behavior. Major and minor phenomena and trends are correctly predicted. The calculated results are judged to agree closely with the data. The calculations will, with few exceptions, lie within the specified or inferred uncertainty bands of the data. The code may be used with confidence in similar applications. The term "major phenomena" refers to phenomena that influence key parameters, such as rod cladding temperature, pressure, differential pressure, mass flow rate, and mass distribution. Predicting the major trends means that the prediction shows the significant features of the data. Significant features include the magnitude of a given parameter through the transient, slopes, and inflection points that mark significant changes in the parameter.

"Reasonable agreement" applies when the code exhibits minor deficiencies. Overall, the code provides an acceptable prediction. All major trends and phenomena are predicted correctly. Differences between calculated values and data are greater than are deemed necessary for excellent agreement. The calculation will frequently lie outside but near the specified or inferred uncertainty bands of the data. However, the correct conclusions about trends and phenomena would be reached if the code were used in similar applications. The code models and/or facility model noding should be reviewed to see if improvements can be made.

"Minimal agreement" applies when the code exhibits significant deficiencies. Overall, the code provides a prediction that is only conditionally acceptable. Some major trends or phenomena are not predicted correctly, and some calculated values lie considerably outside the specified or inferred uncertainty bands of the data. Incorrect conclusions about trends and phenomena may be reached if the code were used in similar applications; an appropriate warning must be issued to users. Selected code models and/or facility model noding must be reviewed, modified, and assessed before the code can be used with confidence in similar applications.

"Insufficient agreement" applies when the code exhibits major deficiencies. The code provides an unacceptable prediction of the test because major trends are not predicted correctly. Most calculated values lie outside the specified or inferred uncertainty bands of the data. Incorrect conclusions about trends and phenomena are probable if the code is used in similar applications; an appropriate warning must be issued to users. Selected code models and/or facility model noding must be reviewed, modified, and assessed before the code can be used with confidence in similar applications.

REFERENCES

A-1. C. D. Fletcher, P. D. Bayless, C. B. Davis, M. G. Ortiz, T. K. Larson, S. M. Sloan, R. A. Shaw, R. R. Schultz, C. E. Slater, G. W. Johnsen, L. S. Ghan, and D. E. Bessette, "Adequacy Evaluation of RELAP5/MOD3, Version 3.2.1.2 for Simulating AP600 Small-Break Loss-of-Coolant Accidents (Final Draft)," Idaho National Engineering Laboratory document INEL-96/0400 (December 1996).

APPENDIX B

ADDITIONAL PERSPECTIVES
SEPARATE EFFECT AND INTEGRAL EFFECT TESTS

Three categories of experimental data have traditionally been identified for use in T-H code validation: separate effect, component effect, and integral effect (Fig. 2-2). The three categories are generally distinguished by both the complexity of the processes/phenomena and the geometric scale of the respective facilities. Separate effect tests (SETs) generally focus on a few processes or phenomena within a single component test fixture, although some multiple component tests are classified as SETs also. Component effect tests (CETs) generally focus on a single component. Integral effect tests (IETs) generally focus on multiple, coupled processes and components in facilities that have numerous hardware components. A gray area arises at the interfaces where assignment of a particular facility or test to the SET, CET, or IET categories is arbitrary.

SET, CET, and IET data are generally applied in different ways within the code development/code qualification process. SET data are most useful for model development. SET data are also the most applicable data for validating flow field models and engineering correlation (closure) and component models.

CET and IET data are most useful for assessing performance and qualifying the integrated T-H code for its targeted applications. IET data can sometimes be used for equipment component model qualification. If sufficient instrumentation is provided in an IET facility, these facilities can assume some characteristics of SET facilities and tests. The SCTF, CCTF, and UPTF facilities have variously been categorized as either SET, CET, or IET facilities, depending upon how they are configured for a given test or test series.

A distinguishing characteristic between SET and IET data is the extent and accuracy of the instrumentation. Instrumentation for SET data can generally have very detailed spatial and temporal resolution and high accuracy. The larger physical scale of integral test facilities generally limits both the spatial and temporal resolution, primarily because of the larger number of instruments and the broader instrumentation ranges to cover the range through which the measured parameter moves during an integral test.

Generally, as experiments move from fundamental separate effect to large-scale integral effect, the situations of interest become more complex, the data become more limited in quality (spatial and temporal resolution and accuracy), interactions between components and physical processes in different components become more important, and understanding of the experimental results becomes much more difficult.

We have attempted to capture the scale and complexity relationships between various SET, CET, and IET facilities in Fig. 2-2 where we show a spectrum of SET, CET, and IET facilities in a matrix. Plant data arising from operational tests, operational transients, and accidents are also shown. The abscissa of the matrix conveys qualitative or semi-quantitative information about facility scale and the ordinate conveys qualitative information about the facility complexity. Within the SET category, separate scales are assigned to fundamental, single component, and several component tests. IET facilities are plotted relative to a volume scale; the positions are approximate.

B.1. Separate Effect Tests

Separate effect experiments are experiments in which a very limited number of physical phenomena are of interest and detailed, high-quality data are obtained. In a steady-state experiment, for example, detailed distributions of pressure, void fraction, and wall temperature will be reported along the flow direction. For the case of transient experiments, instrumentation with temporal resolution sufficient to measure all changes of interest will be employed. The fine spatial and temporal detail and high accuracy of the data make separate effect data appropriate for model development. Predictions of these kinds of experiments usually lead to nearly complete understanding of the code results and resolution of any differences between code predictions and the measured data. In Fig. 2-2, we show three different types of SET facilities: fundamental, single component, and several component.

The objective of fundamental SET facilities is to make a single physical phenomena (e. g., wall friction, momentum flux, gravity, and radiation heat transfer) or some aspect of the numerical solution methods (stability, convergence) dominant in the data. These data are the most prized, then, both for the development of engineering correlations and for evaluating the fundamental models in a T-H code. Frequently, however, it is not possible to isolate a single physical phenomenon. Thus, fundamental tests are also conducted to focus on a single parameter, such as the pressure gradient that arises from the flow process. The two-phase pressure gradient, however, is the integrated result of several fundamental phenomena, e.g., the void distribution both across the flow channel transverse to the flow direction and in the direction of flow, and fluid properties encountered in single-phase flows.

The objective of single-component SETs is broader in that more interacting phenomena and processes occur. Component tests can focus on either the detailed behaviors within the component, e.g., thermal stratification or level changes in a coolant makeup tank; the boundaries of a component, e.g., the output from a circulating pump under a full range of operating conditions; or a combination of both.

Several component tests arise not so much from the desire to combine a few components in a facility but the practical necessity of combining several components to produce the desired test characteristics. In addition, several facilities produce either SET or IET data, depending upon their configuration. Examples are CCTF, SCTF, UPTF, and FLECHT-SEASET.

We view relatively complex physical processes in larger scale facilities to be naturally located near the boundary between separate effect and integral effect experiments. Forced reflood heat transfer of full-length rod bundles is an example of complex separate effect data that generally arises in several component facilities such as FLECHT-SEASET, CCTF, and SCTF when they are operated in a SETs mode.

B.2. Component Effect Tests

CETs investigate behavior in a plant component, frequently but not always at full-scale. Component effect experiments are of several types. Some tests are designed to test the performance and characteristics of a particular component, e.g., a pump or valve. More frequently, however, component data is extracted from an integral test facility that includes several components. The IET facility can be run in an integral mode, component mode, or separate effect mode. The Flecht-Seaset facility is an example of a facility that has utilized this type of flexible design.

B.3. Integral Effect Tests

Integral effect experiments are generally designed to investigate a complete system, or a scaled model of complete nuclear reactor systems. IETs may also be designed to investigate a single phenomena in a complete system, e.g., natural circulation in a complete model of a pressurized water reactor. Finally, IETs frequently develop specific component data, an obvious overlap with some SET facilities.

Generally, the physical scale of the test rigs is such that detailed instrumentation is not possible. Additionally, the data may be difficult to understand, especially as the scale of the facility increases because both the complexity of the physical phenomena and the amount of data taken. Comparison of code predictions with data from these tests may not result in closure of differences between the data and code predictions because of the complexity of both the physical phenomena and the geometry of the region of interest.

Numerous IET facilities simulating nuclear power plants have been designed, built, and operated in the past 30 years. The PWR IETs identified as part of an OECD/CSNI effort to prepare IET data assessment matrices are displayed in Fig. 2-3. The volume scales of the facilities range from 1/1 for UPTF to 1/1705 for Semiscale (see Table 8-2). Similarly, the facility complexity various from the OTIS and GERDA facilities, which were single-loop representations of OTSG PWRs, to LOFT, the only IET facility with a nuclear core.

APPENDIX C

THE MODELS AND METHODS IN TRAC-M

An expanded view of the models and methods in the TRAC-M code is given in the following discussion. The detailed lists developed herein will be used to identify appropriate experimental data for validation of the models and methods.

C.1. Basic-Equation Models

The basic-equation models in TRAC-M were listed in Section 3.1 of the main report. The contents of these model equations are given in more detail in the following paragraphs.

C.1.1. Mass, Momentum, and Energy Equations for the Fluid Flow

The basic fluid flow model equations in TRAC-M are outlined in Sections C.1.1.1 through C.1.1.4 below.

C.1.1.1. Mass Conservation Equations. TRAC-M contains mass conservation equations for

- the liquid phase of water,
- the mixture of the vapor phase of water plus the noncondensable gas,
- noncondensable gases, and
- solids dissolved in the liquid phase.

These equations contain convection and mass exchange contributions. The verification and validation efforts will focus on the mass exchange contribution due to heat transfer, which is a function of specific-area and heat transfer coefficient models.

C.1.1.2. Equations of Motion. TRAC-M contains momentum equations, or equations of motion for

- the liquid phase of water and
- the mixture of vapor and noncondensable gas.

Any solids dissolved in the liquid phase are merely transported by the liquid. There is no feedback from the solids to the liquid equation of motion. This modeling is based on the assumption that the dissolved solids are present in trace amounts in the liquid.

The equations of motion contain accounting of

- momentum flux,
- interfacial drag,
- the pressure gradient,
- momentum exchange due to mass exchange,
- wall-to-phase drag,
- gravity,
- pressure change due to local losses, and
- an area-change contribution.

The local-losses modeling includes abrupt expansion and contraction, turning flow loss, and thin plate orifice.

The wall and interfacial drag contributions contain quantities that are functions of the two-phase flow regime. The verification and validation efforts will consider all the terms in the equations of motion and focus especially on the flow-regime dependent terms. These latter terms are primarily the quantities with the largest uncertainty.

C.1.1.3. Energy Equations. TRAC-M contains energy conservation equations for

- the vapor plus noncondensable gas mixture;
- the liquid-plus-gas vapor mixture, i.e., the entire mixture; and
- the liquid.

The vapor-plus-gas energy equation contains

- energy convection for the mixture of gases,
- a pressure-work contribution,
- wall-to-gas-mixture heat transfer,
- direct energy deposition to the gas mixture by neutrons,
- interface-to-gas-mixture heat transfer, and
- energy exchange due to mass exchange.

The energy equation for the entire mixture contains

- energy convection for the entire mixture,
- a pressure-work contribution,
- wall-to-gas-mixture heat transfer,
- wall-to-liquid heat transfer,
- direct energy deposition to the liquid by neutrons, and
- direct energy deposition to the gas mixture by neutrons.

The energy equation for the liquid contains:

- energy convection for the liquid,
- a pressure-work contribution,
- wall-to-liquid heat transfer,
- direct energy deposition to the liquid by the neutrons,
- heat transfer at the interface, and
- energy exchange due to mass exchange.

As in the case of the equations of motion, the wall-to-phase and interfacial energy exchange will be the focus of the verification and validation efforts for the fluid energy equations. Note that not all the items listed above are unique; some are repeated between the various forms of the energy equations.

The temperature of the liquid and the temperature of the gas mixture, along with the pressure, are the dependent variables for the equation of state in the code.

C.1.1.4. The 3D Vessel Model Equations. The reactor pressure vessel model in TRAC-M contains 3D versions of the fluid flow equations given in the three previous sections above.

C.1.2. Heat Conduction in Solid Structures

The heat conduction model in TRAC is applicable to conduction in rectangular slabs and cylindrical rods. The conduction model includes accounting of

- gap conductance,
- metal-water reaction, and
- temperature and space dependent material properties.

The fuel-clad gap conductance has been found to be important and highly ranked in previous PIRT studies.

There are four numerical solution methods available:

- lumped-parameter (the lumped-capacitance method);
- 1D radial conduction without axial conduction;
- 2D radial plus axial conduction, implicit in the radial direction, and explicit in the axial direction; and
- fully implicit radial and axial conduction for use in reflood modeling. Fine-mesh rezoning is also available for reflood modeling.

C.1.3. Reactor Core Power Model

Three methods are available for calculating the reactor core power in TRAC-M:

- a table as input to the code,
- a point-reactor kinetics model, and
- a 3D neutron kinetics model.

Reactivity feedback is based on changes in

- fuel temperature,
- the coolant temperature,
- coolant void fraction, and
- boron concentration.

C.1.4. Radiative Energy Exchange in the Core

The radiative energy exchange model in TRAC-M accounts for surface-to-surface radiation for solid surfaces that are attached to the same hydrodynamic node. The model also accounts for the effects of a two-phase mixture between the radiating surfaces.

C.1.5. Equations of State

TRAC-M has the following equations of state:

- For the water liquid, the density and specific internal energy are given by functions of the total pressure and the liquid temperature.

- For the water vapor, the density and specific internal energy are given by functions of the partial pressure for the vapor and the gas-mixture temperature.

- For the noncondensable gas, the density and specific internal energy are given by of the partial pressure of the noncondensable gas and the gas mixture temperature.

C.1.6. Other Fluid Properties

The viscosity and thermal conductivity for all fluids in the flow field are also needed. Various derivatives of the equation of state are needed for numerical solution and other purposes.

C.2. Flow Field Models and Engineering Correlations (Closure)

As noted in Section 3.2, closure for the fluid flow equations is based on the use of flow-regime maps plus models and correlations for wall-to-phase and interfacial mass, momentum, and energy exchange. Additional information about the closure for the fluid flow model equations is given in the following discussions.

C.2.1. Flow Regime Map(s)

The flow regime modeling in TRAC includes

- a vertical flow regime map
- a horizontal flow regime map
- modeled flow regimes, including
  single phase
  bubbly
  slug
  annular-mist
  mist
  churn
  horizontal stratified
  vertical stratified

In TRAC-M, the horizontal flow regime map is basically the same as the vertical map.

The flow regime criteria and interfacial area for the individual flow regimes are summarized in Table C-1, which is taken from Reference 3-1. Table C-1 applies to all applications except for reflood heat transfer in the core. Flow regime criteria under reflood conditions are given in Section C.4.

C.2.2. Fluid Mass Equation Closure

Closure of the fluid mass conservation equation models used in TRAC requires accounting of wall-to-phase and interfacial heat transfer and interfacial area to get the mass transfer due to heat transfer. The subcooled boiling model in TRAC-M is part of the closure of the fluid mass balance equations. The solids dissolved in liquid can plate out, and modeling this process is the closure for the dissolved-solids mass conservation equation.

Fluid mass balance equation closure in TRAC-M is summarized in Table C-2, which has been taken from Reference 3-2. Verification, validation, and qualification activities will ultimately be applied to the individual correlations given in the table.

C.2.3. Fluid Momentum Equation Closure

Closure of the fluid equations of motion requires modeling for wall-to-phase and interfacial momentum exchange. Modeling of momentum exchange is needed for both the 1D and 3D equations of motion. The terms in the momentum equations used in TRAC-M have been summarized in Section C.1.1.2. Additional information about the wall and interfacial drag models is given below.

The models and correlations that make up the wall-drag accounting for the equations of motion are summarized in Table C-3. The wall-drag models are used for applications that do and do not involve reflood heat transfer. The interfacial momentum exchange modeling for applications that do not involve reflood heat transfer is summarized in Table C-4. Both Tables have been taken from Reference 3-1. Verification, validation, and qualification activities will ultimately be applied to the individual correlations given in the tables.

C.2.4. Fluid Energy Equation Closure

Closure of the fluid energy equations requires modeling of the wall-to-phase and interfacial energy exchanges. Modeling of the energy exchange is needed for both the 1D and 3D energy equations. The terms in the energy equations used in TRAC-M have been summarized in Section C.1.1.3. Additional information about the wall and interfacial energy exchange models is given below.

The models and correlations that make up the wall-to-phase energy exchange are summarized in Table C-5 for applications that do not involve reflood. The interfacial energy exchange models and correlations for applications that do not use the reflood heat transfer modeling in TRAC are summarized in Table C-6. Verification, validation, and qualification activities will ultimately be applied to the individual correlations given in the tables.

C.3. Equipment Component Models

The system-equipment component models in TRAC-M have been listed in Section C.3. The properties of these models are best determined at present by reference to the TRAC-P Theory Manual.[3-2]

C.4. Special-Purpose Models

The special-purpose models in TRAC-M have been listed in Section 3.4. The special-purpose models that have been found to be important and highly ranked in previous PIRT studies are the (1) CCFL model, (2) critical flow model that determines the flow rate of the fluid under choked-flow conditions, (3) two-phase level-tracking model, and (4) reflood heat transfer model.

The CCFL model in TRAC-M is based on a generalized formulation from which both the Wallis and Kutaladaze forms can be recovered.

The critical flow model in TRAC-M is based on critical flow of (1) a subcooled liquid including modeling of nucleation delay under rapid pressure change conditions, (2) critical flow of a two-phase (liquid and vapor water), two-component (water and a gas) mixture based on the basic fluid flow equations in TRAC, and (3) critical flow based on isentropic expansion of a single-phase vapor.

The reflood heat transfer model in TRAC-M is quite complex and contains special versions of (1) flow-regime modeling, (2) some wall-to-phase energy exchange models, (3) interfacial momentum and energy exchange models and correlations, and (4) special modeling and numerical solution methods for conduction heat transfer. The flow-regime criteria models and correlations are summarized in Table C-7, interfacial momentum exchange models and correlations are summarized in Table C-8, and those for interfacial energy exchange are given in Table C-9. All these tables have been taken from Reference 3-1. Verification, validation, and qualification investigations will ultimately be applied to the individual correlations given in these tables.

The TEE component offtake flow model in TRAC-M includes accounting for three offtake geometries and four offtake flow patterns. The modeling allows calculation of entrainment of liquid and vapor by vapor and liquid, respectively, for example. The control system models and methods may be important for some operational transients. The control system elements in TRAC include

- component hardware actions,
- plant system trips,
- control block functions, and
- use of control system elements for steady state calculations.

The control system elements available in TRAC-M are quite general and can probably model almost any control system encountered in TRAC-M applications.

C.5. Numerical Solution Methods

All the numerical solution methods used in TRAC-M must undergo verification and validation. The solution methods for the fluid flow equations are especially important because they are the bases of almost every analysis done with TRAC. The numerical solution methods associated physical components and phenomena/ processes rated highly important in previous PIRT studies should also receive priority relative to verification and validation.

For completeness of this Section, the numerical solution methods listed in Section 3.5 are repeated here. The solution methods in TRAC-M include those for

- fluid field equations

- 1D SETS method
- 3D SETS method
- conduction in solid materials
  - 1D rectangular and cylindrical
  - 2D rectangular and cylindrical
- power generation in the fuel rods
- the trip and control system elements
- the fluid equation of state
- fluid boundary conditions
- the equipment component models
- the special-purpose models
- steady-state solution methods, and
- timestep size and control methods.

REFERENCES

C-1. M. Ishii and K. Mishima, "Study of Two-Fluid Model and Interfacial Area," Argonne National Laboratory report ANL-80-111 (NUREG/CR-1873) (1980).

C-2. I. Kataoka, M. Ishii, and K. Mishima, "Generation and Size Distribution of Droplet in Annular Two-Phase Flow," *Trans. ASME, J. Fluid Engineering* **105**, 230–238 (1983).

C-3. J. Kitscha and G. Kocamustafaogullari, "Breakup Criteria for Fluid Particles," *Int. J. Multiphase Flow* **15**, 573–588 (1989).

C-4. M. Ishii and K. Mishima, "Droplet Entrainment Correlation in Annular Two-Phase Flow," *Int. J. Heat Mass Transfer* **32**, 1835–1846 (1989).

C-5. K. Mishima and M. Ishii, "Theoretical Prediction of Onset of Horizontal Slug Flow," *Trans. ASME, J. Fluids Engineering* **102**, 441–445 (1980).

C-6. R. T. Lahey, "A Mechanistic Subcooled Boiling Model," *Proc. 6th Int. Heat Transfer Conf.* (Toronto, Canada, 1978), Vol. 1, pp. 293–297.

C-7. P. Saha and N. Zuber, "Point of Net Vapor Generation and Vapor Void Fraction in Subcooled Boiling," *Proc. 5th Int. Heat Transfer Conf.* (Tokyo, Japan, 1974) Paper B4.7.

C-8. S. W. Churchill, "Friction-Factor Equation Spans All Fluid-Flow Regimes," *Chemical Engineering*, pp. 91–92 (Nov. 7, 1977).

C-9. M. Ishii and T. C. Chawla, "Local Drag Laws in Dispersed Two-Phase Flow, "Argonne National Laboratory report ANL-79-105 (NUREG/CR-1230) (December 1979).

C-10. J. F. Lime, "Memorandum on Interfacial Drag from M. Ishii to R. Nelson, July 28, 1987," Los Alamos National Laboratory document LA-UR-01-1591 (March 2001).

C-11. L. Shiller and A. Z. Nauman, *Ver. Deut. Ing.* **77**, 318–320 (1933).

C-12. R. B. Bird, W. E. Stewart, and E. N. Lightfoot, *Transport Phenomena* (John Wiley and Sons, New York, 1960).

C-13. I. Kataoka and M. Ishii, "Mechanism and Correlation of Droplet Entrainment and Deposition in Annular Two-Phase Flow," Argonne National Laboratory report ANL-82-44 (NUREG / CR-2885) (1982).

C-14. G. B. Wallis, *One Dimensional Two Phase Flow* (McGraw-Hill Book Company, New York, 1969).

C-15. Y. Taitel and A. E. Dukler, "A Model for Predicting Flow Regime Transitions in Horizontal and Near Horizontal Gas-Liquid Flow," *AIChE J.* **22** (1), 47–55 (1976).

C-16. A. Ohnuki, H. Adachi, and Y. Murao, "Scale Effects on Countercurrent Gas-Liquid Flow in Horizontal Tube Connected to Inclined Riser," Proceedings of 1987 National Heat Transfer Conference, Pittsburgh, Pennsylvania (August 9-12, 1987), pp. 40–49.

C-17. J. P. Holman, *Heat Transfer*, 5th ed. (McGraw-Hill Book Co., New York, 1981).

C-18. F. E. Incropera and D. P. DeWitt, *Fundamentals of Heat Transfer* (John Wiley & Sons, New York, 1981).

C-19. J. M Delhaye, M. Giot, and M. L. Riethmuller, *Thermohydraulics of Two-Phase Systems for Industrial Design and Nuclear Engineering* (Hemisphere Publishing Corporation, New York, 1981).

C-20. W. M. McAdams, *Heat Transmission*, 3rd ed. (McGraw-Hill Book Co., New York, 1954).

C-21. R. T. Lahey and F. J. Moody, *The Thermal-Hydraulics of a Boiling Water Nuclear Reactor* (American Nuclear Society, La Grange Park, Illinois, 1977). C-22. L. Biasi, G. C. Clerici, S. Garribba, R. Sala, and A. Tozzi, "Studies on Burnout, Part 3: A New Correlation for Round Ducts and Uniform Heating and Its Comparison with World Data," *Energia Nucleare* **14**, 530–536 (1967).

C-23. L. A. Bromley, "Heat Transfer in Stable Film Boiling," *Chem. Eng. Prog.* **46** (5), 221–227 (May 1950).

C-24. S. Webb and J. C. Chen, "A Two-Region Vapor Generation Rate Model for Convective Film Boiling," Int. Workshop of Fundamental Aspects of Post-Dryout Heat Transfer, Salt Lake City, Utah, April 1–4, 1984.

C-25. R. E. Henry, "A Correlation for the Minimum Film Boiling Temperature," *AIChE Symposium Series* **138**, 81–90 (1974).

C-26. M. K. Denham, "Inverted Annular Film Boiling and the Bromley Model," Atomic Energy Establishment Winfrith report AEEW-R 1590 (January 1983).

C-27. F. Kreith, *Principles of Heat Transfer*, 3rd. ed., (Harper and Row, New York, 1973).

C-28. W. M. Rohsenow and H. Choi, *Heat, Mass, and Momentum Transfer* (Prentice-Hall, Englewood Cliffs, New Jersey, 1961).

C-29. Y. M. Chen and F. Mayinger, "Measurement of Heat Transfer at the Phase Interface of Condensing Bubbles," *ANS Proc. 1989 National Heat Transfer Conf.***, (**Philadelphia, Pennsylvania, August 6-9, 1989) HTC-Vol. 4, pp. 147–152.

C-30. S. Whittaker, "Forced Convection Heat Transfer Correlations for Flows in Pipes, Past Flat Plates, Single Cylinder, Single Sphere, and Flows in Packed Beds and Tube Bundles," *AIChE J.* **18**, 361–372 (1972).

C-31. K. Pasamehmetoglu and R. Nelson, "Transient Direct-Contact Condensation on Liquid Droplets," *Nonequilibrium Transport Phenomena* (American Society of Mechanical Engineers, New York, 1987), HTD-Vol. 77, pp. 47–56.

C-32. S. G. Bankoff, "Some Condensation Studies Pertinent to Light Water Safety," *Int. J. Multiphase Flow* **6**, 51–67 (1980).

C-33. G. Ryskin, "Heat and Mass Transfer from a Moving Drop—Some Approximate Relations for the Nusselt Number," *Int. Comm. Heat Mass Transfer* **14**, 741–749 (1987).

C-34. J. H. Linehan, M. Petrick, and M. M. El-Wakil, "The Condensation of Saturated Vapor on a Subcooled Film During Stratified Flow," *Chem. Eng. Symp. Series* **66** (102), 11–20 (1972).

C-35. C. Unal, K. Tuzla, C. A. Tuzla, and J. C. Chen, "Vapor Generation Model for Dispersed Drop Flow," *ANS Proc. 1989 National Heat Transfer Conf.* (August 1989), HTC-Vol. 4.

C-36. G. G. Sklover and M. D. Rodivilin, "Condensation on Water Jets with a Cross Flow of Steam," *Teploenergetika* **23**, 48–51 (1976).

TABLE C-1
TRAC CLOSURE RELATION SUMMARY:
FLOW-REGIME CRITERIA AND INTERFACIAL AREA
FOR NON-REFLOOD APPLICATIONS

Flow Regime	**Flow-Regime Criteria**	**Interfacial Area (A_i)**
Bubbly Flow	$a \leq 0.3$; or $a \leq 0.5$ and $G \geq 2700\ kg/m^2$-s	based on Ishii and Mishima[C-1]
Bubbly Slug Transition	$0.3 < a \leq 0.5$ and $2000 < G < 2700\ kg/m^2$-s	based on Ishii and Mishima[C-1]
Bubbly Slug Flow	$0.3 < a \leq 0.5$ and $G \leq 2000\ kg/m^2$-s	based on Ishii and Mishima[C-1]
Churn Flow	$0.5 < a \leq 0.75$	weighted average of bubbly slug and annular-mist interfacial areas
Annular-Mist Flow	$a > 0.75$	superimpose film and droplet fields; droplet area based on the droplet diameter defined by Kataoka[C-2] or Kitscha and Kocamustafaogullari,[C-3] and on the entrainment fraction of Ishii and Mishima;[C-4] film area based on geometry and entrainment fraction
Transition to Stratified Flow	1D components: gas (or liquid) velocity between 1 and 10 times the critical velocity 3D components: gas velocity between 1 and 2 times the critical velocity	weighted average of stratified flow and basic flow-regime map interfacial areas

TABLE C-1 (cont)
TRAC CLOSURE RELATION SUMMARY:
FLOW-REGIME CRITERIA AND INTERFACIAL AREA
FOR NON-REFLOOD APPLICATIONS

Flow Regime	Flow-Regime Criteria	Interfacial Area (A_i)
Stratified Flow	1D components: critical velocity criteria 3D vessel: horizontal stratified flow uses critical relative velocity of Mishima and Ishii;[C-5] vertical stratified flow uses terminal bubble rise velocity criterion	interfacial area for horizontal stratified flow based on circular pipe geometry; interfacial area for vertical stratified flow based on average cross-sectional area
Plug Flow	liquid side under condensation mode; void fraction (over three contiguous cells) must satisfy plugging criterion	based on circular pipe geometry

TABLE C-2
TRAC CLOSURE RELATION SUMMARY:
INTERFACIAL MASS TRANSFER

Model	Interfacial Mass Transfer
Total Interfacial Mass Transfer Rate (G)	sum of the mass transfer rates from interfacial heat transfer and subcooled boiling
Mass Transfer Caused by Interfacial Heat Transfer (G_i)	based on the sum of the interface-to-gas and interface-to-liquid heat-transfer rates
Mass Transfer Caused by Subcooled Boiling (G_{sub})	based on Lahey's mechanistic model[C-6] for the evaporation fraction and on the modified Saha-Zuber OSV correlation[C-7] (Note: this model is used only when the subcooled boiling heat-transfer coefficient is nonzero)
Plateout of Dissolved Solids	Later

TABLE C-3
TRAC CLOSURE RELATION SUMMARY:
WALL DRAG

Model Type	Wall-to-Liquid Drag Coefficient (c_{wl})	Wall-to-Gas Drag Coefficient (c_{wg})
Single-Phase	single-phase liquid: based on the modified friction factor correlation[C-8] single-phase vapor: zero	single-phase liquid: zero single-phase vapor: based on the modified Churchill friction factor correlation[C-8]
Two-Phase, Homogeneous	based on the modified Churchill friction factor correlation[C-8] using the two-phase mixture Reynolds number	based on the modified Churchill friction factor correlation[C-8] using the two-phase mixture Reynolds number
Two-Phase, Horizontal Stratified	laminar flow: based on fully-developed laminar friction factor relation turbulent flow: based on McAdams friction factor correlation	laminar flow: based on fully-developed laminar friction factor relation turbulent flow: based on McAdams friction factor correlation

TABLE C-4
TRAC CLOSURE RELATION SUMMARY:
INTERFACIAL DRAG FOR NON-REFLOOD APPLICATIONS

Flow Regime	Interfacial Drag Coefficient (c_i)
Bubbly Flow, Bubbly Slug Flow, Bubbly Slug Transition	defined as per Ishii and Chawla[C-9] (bubble diameter and profile slip based on Ishii;[C-10] bubble drag coefficient for three Reynolds number regimes based on Stokes drag law, the empirical relation proposed by Schiller and Nauman,[C-11] and the recommendation of Bird, Stewart, and Lightfoot[C-12])
Churn Flow	weighted average of bubbly slug and annular-mist interfacial drag coefficients
Annular-Mist Flow	based on drift velocity developed by Kataoka and Ishii[C-13] and total interfacial shear force defined as per Ishii and Mishima[C-1] (film interface friction factor obtained from Wallis;[C-14] droplet diameter based on Kataoka, Ishii, and Mishima;[C-2] droplet drag coefficient based on Ishii and Chawla;[C-9] entrainment based on Kataoka and Ishii[C-13])
Transition to Stratified Flow	weighted average of stratified and flow-regime map interfacial drag coefficients
Stratified Flow	derived from the method of Taitel and Dukler[C-15] (interfacial friction factor based on Ohnuki et al.[C-16])
Plug Flow	no specific model for interfacial drag

TABLE C-5
TRAC CLOSURE RELATION SUMMARY:
WALL-TO-FLUID HEAT TRANSFER
FOR BOTH REFLOOD AND NON-REFLOOD APPLICATIONS

Heat-Transfer Regime	Wall-to-Liquid Heat-Transfer Coefficient (h_{wl})	Wall-to-Gas Heat-Transfer Coefficient (h_{wg})
Natural Convection to Liquid	laminar and turbulent natural-convection correlations[C-17]	zero
Forced Convection to Liquid	Dittus-Boelter correlation[C-18]	zero
Nucleate Boiling	based on the total heat flux (as determined by the Chen correlation[C-19]) minus the wall-to-gas heat flux	maximum of either the natural convection[C-20] or Dougall-Rohsenow[C-21] correlations
Critical Heat Flux	Biasi correlation[C-22]	Biasi correlation[C-22]
Transition Boiling	based on the total heat flux minus the wall-to-gas heat flux (the total heat flux is a weighted average of q_{chf}, calculated via Biasi and q_{min}, which is based on natural convection,[C-20] Dougall-Rohsenow,[C-21] modified Bromley,[C-23] and radiation heat-transfer coefficients) reflood model: total heat flux based on exponential decrease from q_{chf} to q_{film}	maximum of either the natural convection[C-20] or Dougall-Rohsenow[C-21] correlations reflood model: Webb-Chen correlation[C-24]
Minimum Stable Film Boiling Temperature	based on the Fauske homogeneous nucleation temperature[C-25]	based on the Fauske homogeneous nucleation temperature[C-25]

TABLE C-5 (cont)
TRAC CLOSURE RELATION SUMMARY:
WALL-TO-FLUID HEAT TRANSFER
FOR BOTH REFLOOD AND NON-REFLOOD APPLICATIONS

Heat-Transfer Regime	Wall-to-Liquid Heat-Transfer Coefficient (h_{wl})	Wall-to-Gas Heat-Transfer Coefficient (h_{wg})
Film Boiling	based on the modified Bromley film boiling heat-transfer coefficient[C-23] and a radiation term reflood model: based on the Denham[C-26] and modified Bromley[C-23] correlations and a radiation term	maximum of either the natural convection[C-20] or Dougall-Rohsenow[C-21] correlations reflood model: based on Webb-Chen correlation[C-24]
Single-Phase Vapor	zero	maximum of the turbulent natural-convection correlation and either the Sieder-Tate[C-12] or Dittus-Boelter[C-18] correlations
Condensation	zero or the maximum of the laminar natural-convection, turbulent natural-convection, and Chen[C-19] (S = 0) correlations	based on Nusselt, turbulent natural-convection[C-27] and turbulent forced-convection[C-17] correlations
Two-Phase Forced Convection	maximum of the Rohsenow-Choi[C-28] and Dittus-Boelter[C-18] correlations	zero or the maximum of the turbulent natural-convection[C-17] and Dittus-Boelter[C-18] correlations

TABLE C-6
TRAC CLOSURE RELATION SUMMARY:
INTERFACIAL HEAT TRANSFER
FOR NON-REFLOOD APPLICATIONS

Flow Regime	Interface-to-Liquid Heat-Transfer Coefficient (h_{il})	Interface-to-Gas Heat-Transfer Coefficient (h_{ig})	Liquid-to-Gas Sensible Heat-Transfer Coefficient (h_{gl})
Bubbly Flow, Bubbly Slug Flow, Bubbly Slug Transition	condensation or evaporation: based on the Chen and Mayinger[C-29] and the Whittaker[C-30] Nusselt number correlations flashing: based on liquid superheat subcooled boiling: h_{il} is weighted to include Lahey and Moody model[C-21]	1000 W / m^2-K	1000 W / m^2-K
Churn Flow	cond / evap: based on weighted average of annular-mist and bubbly slug heat-transfer factors flashing: based on maximum of weighted heat-transfer factor and liquid superheat relation	based on weighted average of annular-mist and bubbly slug heat-transfer factors	based on weighted average of annular-mist and bubbly slug heat-transfer factors

TABLE C-6 (cont)
TRAC CLOSURE RELATION SUMMARY:
INTERFACIAL HEAT TRANSFER
FOR NON-REFLOOD APPLICATIONS

Flow Regime	Interface-to-Liquid Heat-Transfer Coefficient (h_{il})	Interface-to-Gas Heat-Transfer Coefficient (h_{ig})	Liquid-to-Gas Sensible Heat-Transfer Coefficient (h_{gl})
Annular-Mist Flow	cond/evap: superimpose droplet and film field droplet field: based on transient conduction solution[C-31] film field: based on Bankoff correlation for Stanton number[C-32] flashing: based on maximum of weighted heat-transfer factor and liquid superheat relation	superimpose droplet and film field droplet field: based on Ryskin correlation for Nusselt number[C-33] film field: based on Bankoff correlation for Stanton number[C-32]	superimpose droplet and film field droplet field: based on Ryskin correlation for Nusselt number[C-33] film field: based on Bankoff correlation for Stanton number[C-32]
Transition to Stratified Flow	cond/evap: weighted average of stratified and flow-regime map heat-transfer factors flashing: based on maximum of weighted heat-transfer factor and liquid superheat relation	heat-transfer factor equivalent to value calculated from basic flow-regime map	heat-transfer factor equivalent to value calculated from basic flow-regime map

TABLE C-6 (cont)
TRAC CLOSURE RELATION SUMMARY:
INTERFACIAL HEAT TRANSFER
FOR NON-REFLOOD APPLICATIONS

Flow Regime	Interface-to-Liquid Heat-Transfer Coefficient (h_{il})	Interface-to-Gas Heat-Transfer Coefficient (h_{ig})	Liquid-to-Gas Sensible Heat-Transfer Coefficient (h_{gl})
Stratified Flow	cond / evap: based on Linehan Stanton number relation[C-34] flashing: based on maximum of weighted heat-transfer factor and liquid superheat relation	heat-transfer factor equivalent to value calculated from basic flow-regime map	heat-transfer factor equivalent to value calculated from basic flow-regime map
Plug Flow	condensation: weighted average of flow-regime map, stratified, and plug-flow heat-transfer factors (plug-flow HTC is calculated from a constant Stanton number model)	heat-transfer factor equivalent to value calculated from basic flow-regime map	heat-transfer factor equivalent to value calculated from basic flow-regime map

TABLE C-7
TRAC CLOSURE RELATION SUMMARY:
FLOW-REGIME CRITERIA AND INTERFACIAL AREA
FOR REFLOOD APPLICATIONS

Flow Regime	Flow-Regime Criteria	Interfacial Area (A_i)
Reflood: Bubbly Flow	transition to IAF defined by mechanistic elevation model based on critical heat flux, film-boiling heat flux, and void fraction	defined as above
IAF	flow regime defined by mechanistic elevation models based on capillary number and limited by a range of void fractions	based on liquid core geometry
Dispersed Flow	flow regime defined by mechanistic elevation model based on capillary number and limited by a range of void fractions	superimpose droplet and film fields (similar to annular-mist flow regime); droplet area based on the droplet diameter defined by Kataoka[C-2] or Kitscha and Kocamustafaogullari;[C-3] film area based on geometry and the stable liquid film thickness
Low-Velocity, Vertical Flow	1D components; inclination ≥45 degrees; liquid temperature greater than saturated vapor temperature; gas velocity <0.1 m / s; maximum void fraction over three contiguous cells >0.50; cell void fraction <0.999	based on average cross-sectional area

TABLE C-8
TRAC CLOSURE RELATION SUMMARY:
INTERFACIAL DRAG MODELS
FOR REFLOOD APPLICATIONS

Flow Regime	Interfacial Drag Coefficient (c_i)
Reflood: Subcooled Boiling	composed of the drag coefficient from bubbles at the wall (based on the Colebrook turbulent friction factor) and by the drag coefficient from free-stream bubbles (based on Ishii[C-10])
Smooth IAF	based on smooth tube friction factor correlations (laminar and turbulent flow)
Rough-Wavy IAF	based on Colebrook friction factor for rough walls (relative roughness based on Ishii entrained droplet diameter[C-10])
Agitated IAF	same as rough-wavy IAF
Post-Agitated (Dispersed) Flow	weighted average of agitated IAF and highly dispersed interfacial drag coefficients
Highly Dispersed Flow	composed of separate droplet and film terms; droplet interfacial drag based on form drag of Ishii and Chawla[C-9] and on Ishii[C-10] droplet size; film interfacial drag based on modified Wallis friction factor (film thickness derived by Pasamehmetoglu[C-17])
Low Velocity, Vertical Flow	no specific model for interfacial drag

TABLE C-9
TRAC CLOSURE RELATION SUMMARY: INTERFACIAL HEAT TRANSFER FOR REFLOOD APPLICATIONS

Flow Regime	**Interface-to-Liquid Heat-Transfer Coefficient (h_{il})**	**Interface-to-Gas Heat-Transfer Coefficient (h_{ig})**	**Liquid-to-Gas Sensible Heat-Transfer Coefficient (h_{gl})**
Reflood: Bubbly Flow	cond/evap: weighted average of bubbly, IAF, and dispersed flow heat-transfer factors defined as above, this table	weighted average of bubbly, IAF, and dispersed flow heat-transfer factors defined as above, this table	weighted average of bubbly, IAF, and dispersed flow heat-transfer factors defined as above, this table
IAF	based on HTVSSL model for subcooled liquid kinetic theory of evaporation for flashing[C-35]	$3x10^3$ W/m^2-K	10^3 W/m^2-K
Dispersed Flow	heat-transfer factor equivalent to IAF value flashing: based on maximum of above evap/cond factor and liquid superheat relation	based on Unal[C-35] model	weighted average of Ryskin[C-33] and Bankoff[C-32] models

TABLE C-9 (cont)
TRAC CLOSURE RELATION SUMMARY:
INTERFACIAL HEAT TRANSFER
FOR REFLOOD APPLICATIONS

Flow Regime	Interface-to-Liquid Heat-Transfer Coefficient (h_{il})	Interface-to-Gas Heat-Transfer Coefficient (h_{ig})	Liquid-to-Gas Sensible Heat-Transfer Coefficient (h_{gl})
Low Velocity, Vertical Flow	weighted average of flow-regime map and low velocity, vertical flow heat-transfer factors (vertical flow factor based on pressurizer data assessment)	weighted average of flow-regime map and low velocity, vertical flow heat-transfer factors (vertical flow factor based on kinetic gas theory)	no modification
Effect of Noncondensables	evaporation: heat-transfer factor calculated by flow-regime-independent diffusion model condensation: heat-transfer factor adjusted using model of Sklover and Rodivilin[C-36]	no modification	no modification

APPENDIX D

PIRT PLANT AND SCENARIO DESCRIPTIONS

The PIRT library used to formulate the consolidated list of highly ranked PIRT processes/phenomena was discussed in Section 4.2. In this appendix, brief descriptions of each plant and accident scenario used in preparing the consolidated PIRT are provided.

D.1. Westinghouse AP600 LBLOCA (PWR)

D.1.1. Plant Description

As described in Ref. D-1, the AP600 is a two-loop design. Each loop contains one hot leg, one steam generator, two reactor coolant pumps, and two cold legs. A pressurizer is attached to one of the hot legs. The reactor coolant pumps are a canned-motor design and are attached directly to the steam generator. The loop seal is eliminated; an added safety feature in that core uncovery caused by the existence of water-filled loop seals is eliminated during a postulated small-break LOCA. The core is designed for a low power density and consists of 145 fuel assemblies with an active fuel length of 12 ft. The fuel assembly is a 17 x 17 array of fuel and control rods.

The AP600 incorporates passive safety systems that rely only on redundant and fail-safe valves, gravity, natural circulation, and compressed gas. There are no pumps, diesels, or other active machinery in these safety systems. During plant shutdown, all the passive safety features will be tested to demonstrate system readiness, flow, and heat removal performance. These systems are shown in an isometric cutaway view of the AP600 reactor design in Fig. D-1. Two Passive Safety Injection System (PSIS) trains, each with an accumulator, a Core Makeup Tank (CMT), and an injection line from the In-containment Refueling Water Storage Tank (IRWST) and sump are connected directly to the reactor-vessel downcomer via a direct vessel injection line.

Depressurization of the primary system is an essential process that is required to ensure long-term cooling of the AP600. For example, the accumulators inject coolant into the reactor coolant system only after the primary pressure has dropped to 700 psia. Coolant injection from large, safety-class water pools, specifically the IRWST and sump, can occur only after the reactor coolant system pressure decreases below the gravitational head of each pool. An Automatic Depressurization System (ADS) permits a controlled pressure reduction of the reactor coolant system. The ADS valves open in stages, based upon either reductions in CMT levels to a specified setpoint or elapsed time from a designated event.

After the accumulators and CMTs are depleted and the primary system has depressurized and approached the containment pressure, water injection is provided from the IRWST. This tank empties after several days. Provisions are also made for recirculating coolant from a sump. IRWST and sump recirculation may occur at the same time for some transients.

The AP600 containment plays an essential role in the long-term cooling of the primary via the Passive Containment Cooling System. Steam entering the containment, either through a break in the primary or through operation of the ADS, condenses on the inside of the steel containment shell. The condensate drains downward and a large fraction is delivered via gutters to either the IRWST or the sump. Heat transfer on the outside of the containment steel shell is by evaporation of liquid sprayed near the top of the steel reactor containment dome by the Passive Containment Cooling System, and by convection to an air stream induced by buoyancy-driven flow (unforced).

D.1.2. Scenario Description

As described in Ref. D-1, the LBLOCA scenario is subdivided into four time periods that characterize events during the sequence. These time periods, termed blowdown, refill, reflood, and long-term cooling are defined by the core and lower-plenum liquid-mass-fraction behaviors; the first three periods are shown in Fig. D-2. The scenario description that follows is largely based upon a TRAC-PF1/MOD2 calculation of an 80% DEGB in a single cold-leg pipe between the primary coolant pump and the connecting point for the CMT pressure balance line to the cold leg.[D-2]

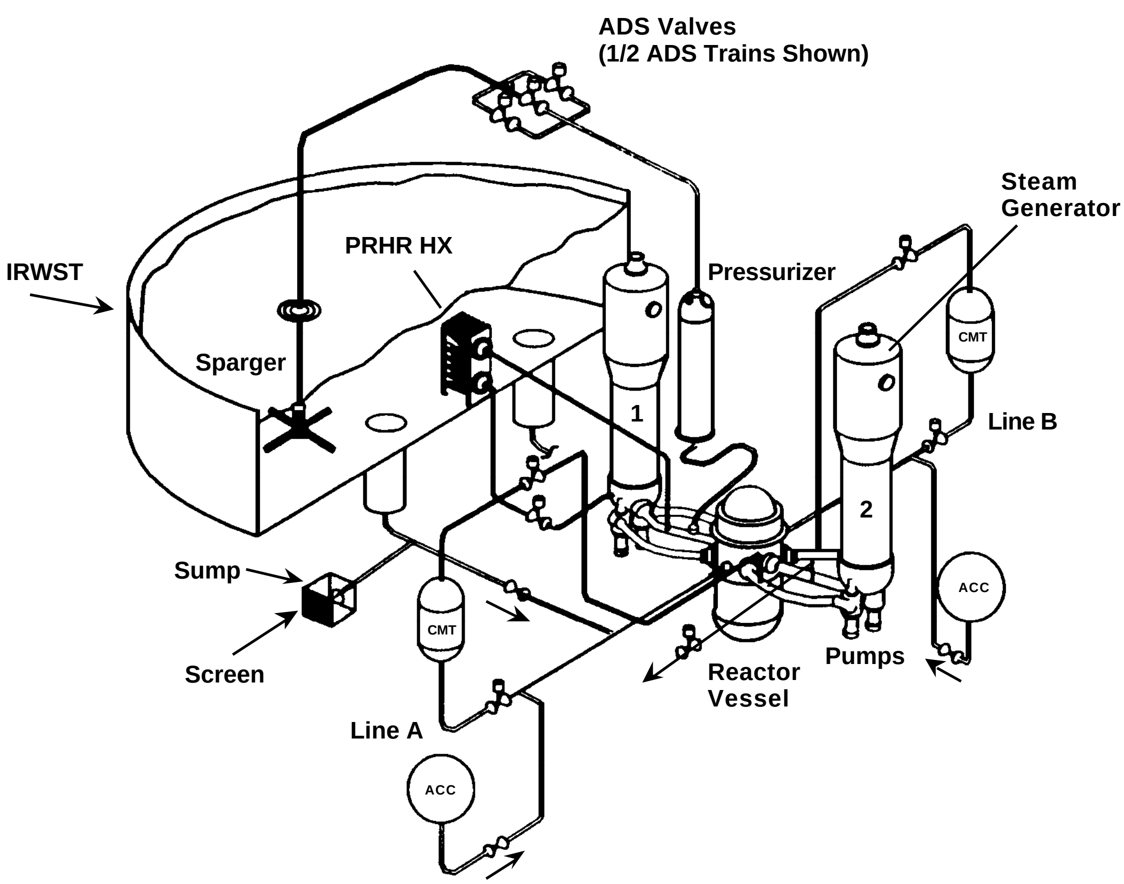


Fig. D-1. AP600 passive safety systems.

The *blowdown period* is the result of a break in the coolant system through which the primary coolant is expelled. Early blowdown physical phenomena include critical flow at the break, fluid flashing and depressurization, redistribution of fuel rod stored energy, and heating of the fuel rods due to degraded heat transfer. Later in the blowdown coolant reenters the core when the intact loop pump flows briefly exceed the break flows. Coolant also drains into the core during this period from the upper plenum. During blowdown, some components are affected more than others. In particular, the heat removal from the core results from the changing flow and heat transfer regimes in the core. The performance of the primary coolant pumps degrades as the coolant flashes. The steam generator heat transfer degrades after the steam-generator secondaries are isolated. The blowdown period ends when the intact-loop accumulator injection is initiated.

During *the refill period*, the reactor system starts to recover as the PSIS components (CMTs and accumulators) start to inject coolant into the primary system. The important refill components and phenomena concern the introduction of water into the reactor vessel downcomer and its subsequent distribution. Refill physical phenomena are the operation of the PSIS, including interactions between the accumulators and CMTs, bypassing injected water through the downcomer to the broken cold leg, and penetration of safety injection water into the lower plenum. The refill period ends when the mixture level in the lower plenum approaches the core inlet, and conditions are established for reflooding the core with coolant.

The *reflood period* begins once the lower plenum has refilled and the core liquid inventory enters a period of sustained recovery. The reflood process is highly oscillatory after the downcomer fills to the direct vessel injection line nozzle but the

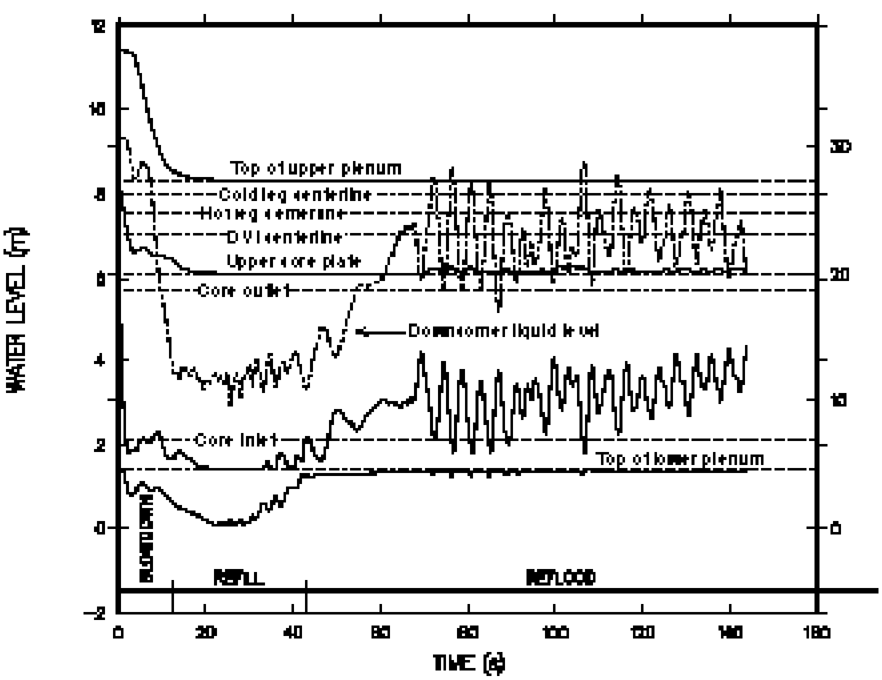


Fig. D-2. Vessel liquid volume fractions.

overall trend with increasing time is increasing core coolant inventory, i.e., a sustained recovery. Refilling of the core with coolant is well advanced by the end of the period. The reflood processes may be quite slow because much of the water is boiled and transported as steam and entrained droplets into the upper plenum and hot-leg piping. The reflood period ends when the entire core is quenched, that is, all fuel rod cladding temperatures are at or slightly above the coolant saturation temperature.

The *long-term cooling period* continues after the entire core quenches. At the time the fuel rod cladding is completely quenched, the core is only partially full. Accumulator discharge is still underway. After the accumulators empty, the CMTs resume draining their inventory into the primary. CMT draining leads to ADS actuation. IRWST injection is initiated when the primary pressure decreases to a level less than the static head in the IRWST. CMT and IRWST draining may occur simultaneously. Draining of the IRWST is expected to take several days, after which water from the sump is recirculated indefinitely. ADS stages 1–3 have an insignificant impact on the transient because the primary has largely depressurized to containment conditions before they open. After the inventory in one of the CMTs drops to 20% of its initial value, fourth stage ADS opens a direct path for release of core-generated steam to the containment. For many accident scenarios, the depressurization process must be assisted by operation of the ADS. However, the LBLOCA has sufficient area to depressurize the primary, even in the absence of ADS actuation.

D.2. Westinghouse 4-Loop Plant SBLOCA (PWR)

D.2.1. Plant Description

The following description is for the Callaway nuclear power plant.[D-3] The Westinghouse 4-Loop SBLOCA PIRT was based upon the Indian Point Unit 2 plant[D-4] but a description of that plant was not readily available.

The nuclear steam supply system consists of a reactor and four closed reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator (Fig. D-3). The nuclear steam supply system also contains an electrically heated pressurizer.

High-pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generators. Here it gives up its thermal energy to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel. The entire reactor coolant system is composed of leaktight components to ensure all fluids are confined to the system.

The reactor coolant pumps are Westinghouse vertical, single-stage, centrifugal pumps of the shaft seal type.

The steam generators are Westinghouse Model F vertical U–tube units that contain thermally treated Inconel tubes.

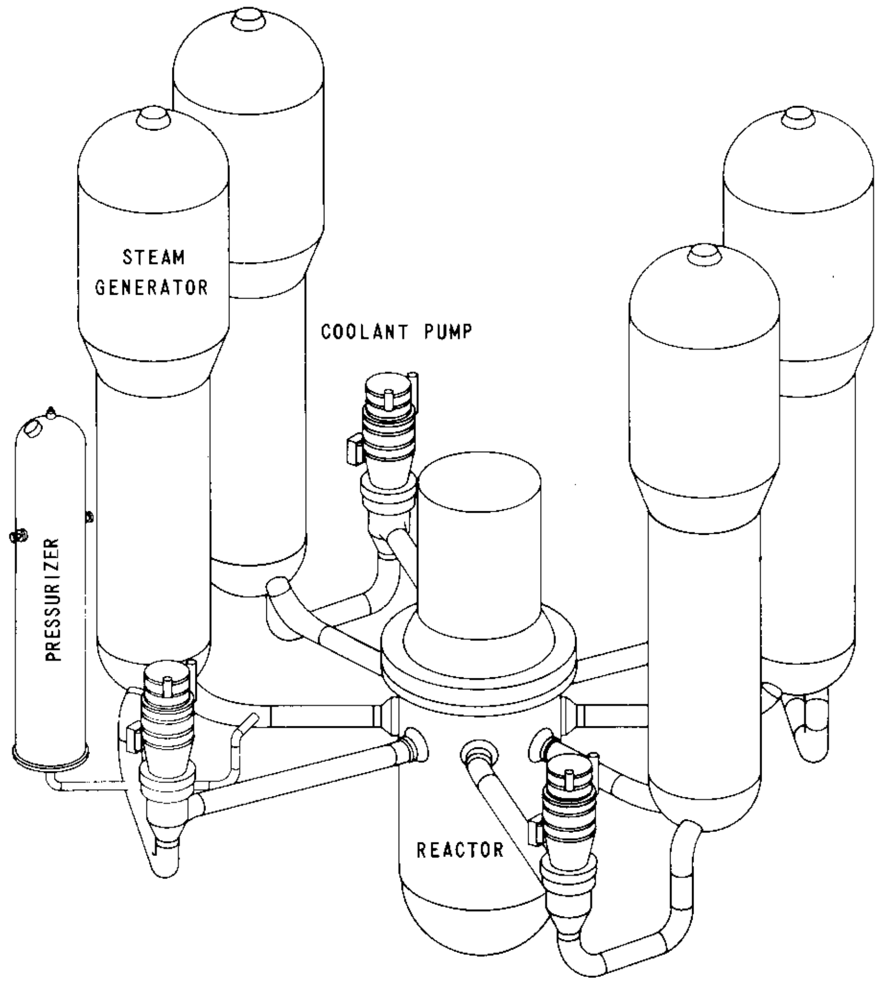


Fig. D-3. Simplified diagram of Westinghouse 4-loop nuclear steam supply system.

Essentially all of the metal surfaces in contact with the reactor water are stainless steel, except the steam generator tubes and the fuel rods which are Inconel and Zircaloy respectively.

An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation and limits pressure variations during plant load transients.

The ECCS injects borated water into the reactor coolant system following a LOCA. This limits damage to the fuel assemblies and limits metal–water reactions and fission product release. The ECCS also provides continuous long-term post-LOCA cooling of the core by recirculating borated water between the containment sumps and the reactor core.

D.2.2. Scenario Description

As described in Ref. D-4, the small-break transient is characterized by five periods: blowdown, natural circulation, loop seal clearance, boil-off, and core recovery. While the duration of each period is break-size-dependent, the small LOCA transient can be characterized as follows:

Blowdown: On initiation of the break, there is a rapid depressurization of the primary side of the reactor cooling system. Reactor trip is initiated on a low pressurizer pressure setpoint. Pump trip occurs either automatically at reactor trip (if the assumption is made that off-site power is lost coincident with reactor trip), or by the operators approximately 15-45 seconds following reactor trip if offsite power is available, based on plant Emergency Operating Procedures. Loss of condenser steam dump effectively isolates the steam generator secondary side, causing it to pressurize to the safety valve setpoints, and release steam through the safety valves. A safety injection signal occurs when the primary pressure decreases below the pressurizer low-low pressure setpoint and safety injection begins after a signal delay time. The reactor cooling system remains liquid solid for most of the blowdown period, with phase separation starting to occur in the upper head, upper plenum and hot legs near the end of this period. During the blowdown period, the break flow is single phase liquid only. Eventually, the rapid depressurization ends and the RCS reaches a pressure just above the steam-generator secondary-side pressure.

Natural Circulation: At the end of the blowdown period, the reactor cooling system reaches a quasi-equilibrium condition that can last for several hundred seconds, depending upon break size. During this period, the loops seals remain plugged and the system drains from the top down with voids beginning to form at the top of the steam generator tubes and continuing to form in the upper head and top of the upper plenum region. The steam generators remove decay heat during this time. Vapor generated in the core is trapped with the reactor cooling system by liquid plugs in the loop seals, and a low quality flow exits the break.

Loop Seal Clearance: The third period is the loop seal clearance period. When the liquid level in the downhill side of the steam generator is depressed to the elevation of the loop seal, steam previously trapped in the reactor cooling system can be vented to the break. The break flow, previously a low-quality mixture, transitions primarily to steam. Prior to loop seal venting, the inner vessel mixture level can drop rapidly, resulting in a deep but short core uncovery. Following loop seal venting, the core level recovers to about the cold leg elevation, as pressure imbalances throughout the reactor cooling system are relieved.

Boil-Off: Following loop seal venting, the vessel mixture level will decrease. In this period, the decrease is due to the gradual boil-off of the liquid inventory in the reactor vessel. The mixture level will reach a minimum, in some cases resulting in a deep core uncovery. The boil-off period ends when the core collapsed liquid level reaches a minimum. At this time, the reactor cooling system has depressurized to the accumulator setpoint, and the core boil-off rate matches the delivery of safety-injection to the vessel.

Core Recovery: The core recovery period extends from the time at which the inner vessel mixture level reaches a minimum in the boil-off period until all parts of the core quench and are covered by a low-quality mixture. The small-break LOCA is considered over, and the calculation is terminated once the entire core is quenched and the safety injection flow exceeds the break flow.

D.3. Babcock & Wilcox 2-x-4 Plant SBLOCA (PWR)

D.3.1. Plant Description

As described in Ref. D-5, the plant selected for the PIRT effort was a typical B&W lowered loop design (Fig. D-4). This design features two hot legs and four cold legs. The elevation of the lowest part of the cold leg is about 6 ft lower than the bottom of the reactor vessel, hence the name “lowered loop.” The reactor coolant pumps are mounted such that the centerline of the discharge is 3.5 ft higher that the reactor vessel inlet piping. A section of the cold leg has an upward slope of 45 degrees to make up the elevation difference. One high-pressure injection line is connected to each of the cold legs on the side of this sloped section so that gravity will direct the high-pressure injection flow toward the reactor vessel.

A unique feature of the B&W vessel internals is the reactor vessel vent valves. These are circular flapper valves, hinged at the top, and are in the closed position held by gravity during normal operation. Eight of these valves are situated around the perimeter of the upper part of the downcomer and allow flow from the upper plenum to the downcomer. If the pressure in the upper plenum increases 0.1 psi greater than the pressure in the downcomer, the valves start to swing open, allowing mass flow from the upper plenum into the downcomer. The reactor vessel vent valves are fully open at 0.25 psid. Thus, the reactor vessel vent valves limit the possibility of pressure building in the upper plenum and depressing the core level.

The two steam generators of B&W design are once through, counter-current-flow heat exchangers. The primary coolant flows vertically downwards, between two plenums, through about 15,500 52-ft-long tubes. Because the primary coolant enters the steam generators at the top, the hot leg must rise up past the top of the steam generators and bend down to connect to the upper plenum. The characteristic inverted U–bend shape gives the hot lag a candy cane appearance. The uppermost part of this hot leg U–bend is a potential location for accumulation of vapor that can block the primary flow path. If the hot leg should drain such that the level falls below the U–bend, primary coolant flow will be interrupted. The U–bend has a small vent valve that can be opened by the operator to vent any bubbles that may have collected at the top.

In the secondary side of the steam generators, subcooled feedwater, preheated before it enters the steam generator, comes in through several nozzles located around the perimeter of the generator about midway between the bottom and top. The feedwater flows through an annular downcomer to the lower plenum and upward through the center of the steam generator, on the outside of the tubes. As the feedwater enters the downcomer, it mixes with saturated steam, which is pulled from the center of the steam generator through an aspirator. Sufficient steam mixes with the feedwater to produce saturated water at the bottom of the downcomer. Once in its upward path, the water

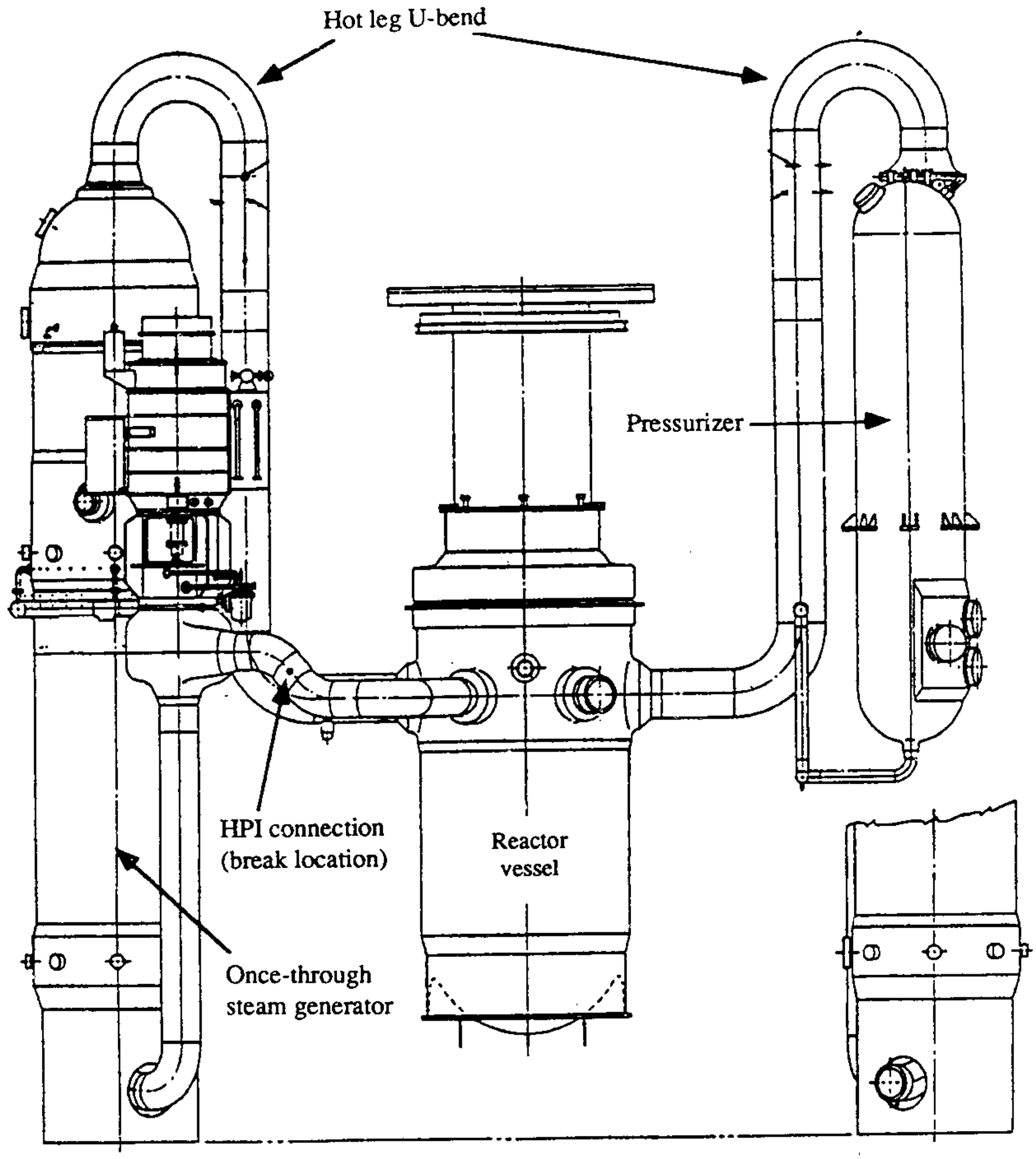


Fig. D-4. Typical B&W lowered-loop plant design.

boils and the generated steam superheats. As the water flows through the tube bundle, it is converted to steam, so that at the level of the aspirator, all of the liquid has been converted to saturated steam. The length of tubes remaining between the aspirator and the upper tube sheet then serve to superheat the steam. Steam superheated to about 33 K (60°F) leaves the generator through the steam annulus and into the steam line.

D.3.2. Scenario Description

As described in Ref. D-6, the SBLOCA scenario is subdivided into four time periods that characterize events during the sequence. The four time periods of the scenario are the following (Fig. D-5).

Blowdown Period: This phase begins with break initiation and ends with the end of reactor coolant pump coastdown. Following break initiation, the reactor begins a fast depressurization, which triggers the reactor trip. It is expected that flashing will start occurring throughout the hot path of the primary, as the primary begins to lose its subcooling margin. If sufficient flashing occurs, the depressurization may slow somewhat before the reactor trip occurs. Once the reactor trips, the heat source decreases rapidly and the rate of depressurization again increases. The operator becomes aware of the loss of subcooling margin and trips the reactor coolant pumps, as established by the emergency operating procedures for this plant. It is expected that at the end of this phase most of the primary side is single phase, conditions approach saturation, and the pump coastdown ends.

Saturation–Natural Circulation Period: This phase begins at the end of the pump coastdown and ends with the complete loss of natural circulation. The subcooling margin has been lost at the middle of this phase and the pressure remains on a plateau (saturation pressure) during this phase. The flow is becoming two-phase and a bubble begins to form at the top of the candy cane. As more and more steam is generated, it

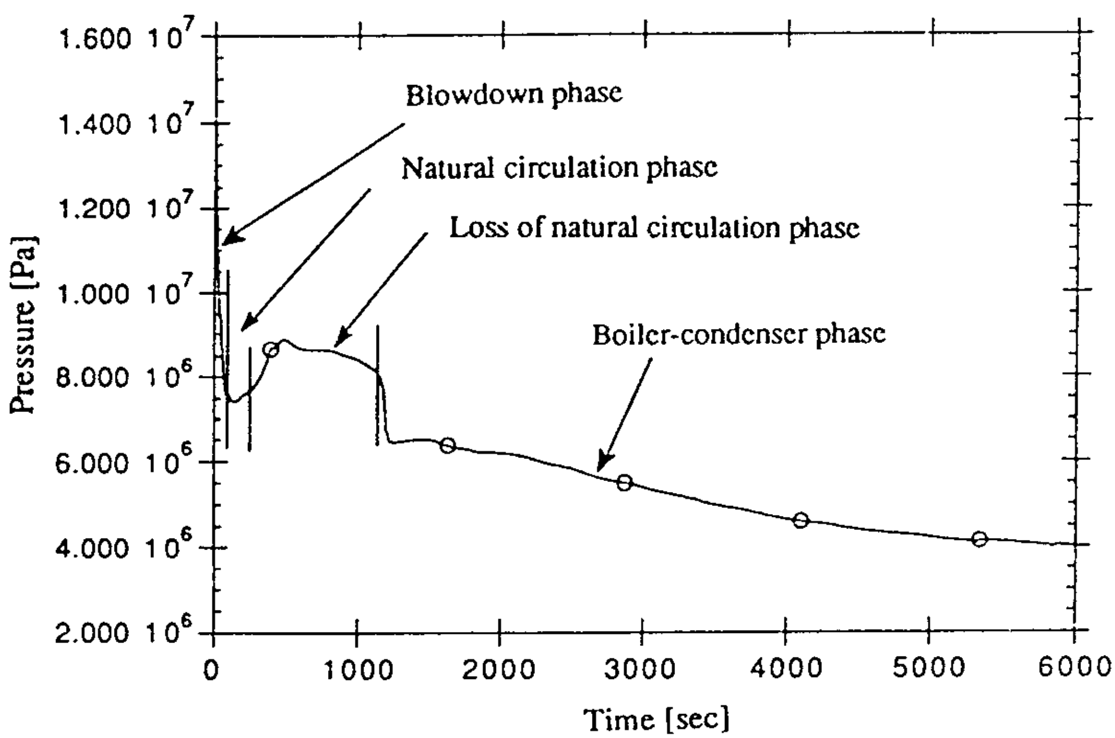


Fig. D-5. Scenario phases for B&W SBLOCA scenario.

becomes increasingly difficult for the natural circulation flow to sweep away the bubbles that now tend to accumulate at the top of the candy cane. A short intermittent mode is expected as the steam accumulates and the two-phase level recedes downward in the candy cane, thus momentarily interrupting the natural circulation. Once natural circulation is interrupted, the loss of the secondary heat sink results in repressurization of the primary. The pressure increase will compress the bubble on top of the candy cane, reestablishing the natural circulation. After a few cycles, the bubble will become too large to allow the liquid to rise to the inverted U-bend and the natural circulation will be interrupted permanently, thus ending this phase.

Loss of Natural Circulation Period: This phase begins with the loss of natural circulation and ends when the vessel steam begins to enter the candy cane. Having lost natural circulation, the pressure begins to increase once again. The main cooling mechanism for the core becomes internal vessel circulation. The reactor vessel vent valves open a flow path that allows the core outlet fluid into the downcomer where it can mix with the incoming high-pressure-injection coolant and recirculate through the core or communicate with the break. During the loss of natural circulation, the operator may decide to sequentially start, run, and stop the reactor coolant pumps, i.e., bump the pumps according to the emergency operating procedures. If the reactor coolant pumps are not bumped, the transient will eventually develop into the next phase, the boiler-condenser mode phase. The steam from the upper plenum begins to flow through the hot leg and find a condensing surface in the steam generator, thus removing decay heat.

Boiler-Condenser Mode Period: In this phase, the steam generated in the core condenses in the primary-side surface of the steam generator tubes and the secondary heat sink is reestablished. The pressure will drop as energy is removed by the boiler-condenser mode and through the break. This phase ends when ECCS begins to refill the primary and the plant enters a recovery phase.

D.3. General Electric LBLOCA (BWR/4)

D.3.1. Plant Description

A simplified diagram of the BWR/4 system configuration is presented in Fig. D-6. As described in Ref. D-7, the principal components of a BWR/4 system include the following.

- Reactor Vessel and Internals: Reactor pressure vessel, jet pumps, steam separators and dryers, core, and core support structures.
- Reactor Water Recirculation System: Pumps, valves, and piping used in providing and controlling flow.
- Main Steam Lines: Main steam valves, piping and pipe supports from reactor pressure vessel up to and including the isolation valves outside of the primary containment barrier.
- Control Rod Drive System: Control rods, control rod drive mechanisms and hydraulic system for insertion and withdrawal of the control rods.

- Nuclear Fuel and Instruments: The nuclear fuel (7 x 7) is located inside the core shroud.
- Engineering Safety Features: Pumps, valves, piping and water storage used to provide cooling and system inventory replacement during accident conditions. High Pressure Coolant Injection (HPCI), Low Pressure Core Spray (LPCS), Low Pressure Core Injection (LPCI), Automatic Depressurization System (ADS), and Residual Heat Removal (RHR).

The Reactor Vessel is divided into five regions: Lower Plenum, Core, Upper Plenum, Dome, and Downcomer region.

There are two external recirculation pumps and 20 internal jet pumps. Each recirculation line feeds five pairs of jet pumps, which are located outside the core shroud throughout

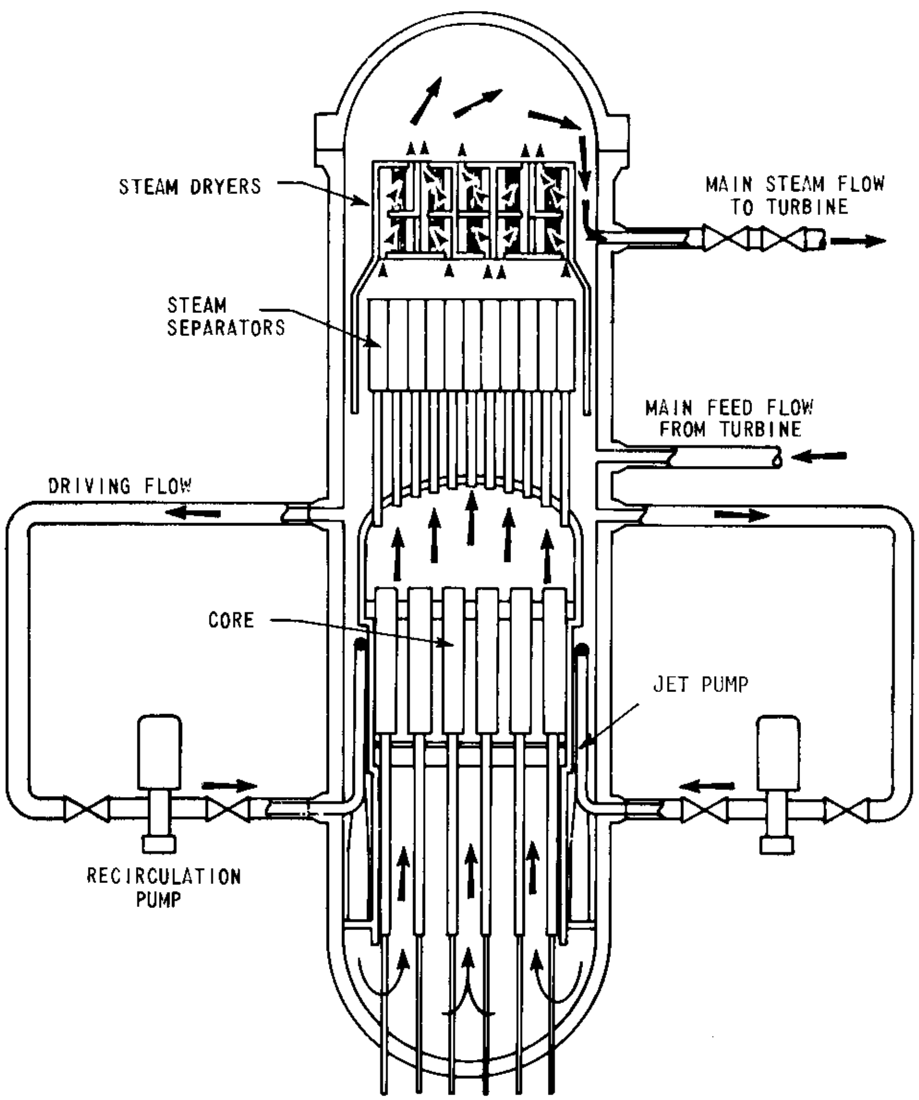


Fig. D-6. Simplified BWR/4 system illustration.

the perimeter of the reactor vessel. The jet pumps provide approximately two-thirds of the recirculation flow within the reactor vessel. Approximately one-third of the core flow is taken from the vessel through the two external recirculation loops. The external loop flow is pumped at a higher pressure, distributed through a manifold to which a number of riser pipes are connected, and returned to the vessel inlet nozzles. This flow is discharged from the jet pump throat where, due to a momentum exchange process, it induces the surrounding water in the downcomer region to be drawn into the jet pump throat. The two flows mix and then diffuse in the diffuser to be finally discharged into the lower plenum.

The BWR/4 power level is 3359 MWt, with a core consisting of 764 fuel assemblies. The steam separator and dryers are located above the core. This equipment is utilized to separate the steam from the liquid in order to avoid excessive rates of liquid in the steam supply system.

The control rods are utilized to effectively and rapidly reduce the power by absorption of neutrons. They are inserted from the bottom of the reactor vessel. There is one control rod for every four fuel assemblies in the core.

The ECCS for a BWR/4 consists of high-pressure coolant injection (HPCI), a low-pressure core spray system (LPCS), a low-pressure coolant injection system (LPCI), and an automatic depressurization system (ADS). The HPCI consists of a single motor driven pump and is designed to inject water into the vessel over the full range of operating pressures. The HPCI uses the condensate storage tank as the initial water supply and upon exhaustion of this source, the suppression pool provides water to this spray system. The injected coolant is injected into the vessel downcomer through the feedwater line.

The LPCS is a low-pressure core spray system. This low-pressure spray system is designed to provide injection for the larger breaks that result in rapid depressurizations of the vessel. The LPCS is also injected into the upper plenum through a circular sparger around the periphery of the core. The function of the LPCS is to limit the peak clad temperatures for intermediate and large breaks, whereas HPCI, along with ADS, is intended for core cooling following small breaks. The LPCS draws water from the suppression pool. The LPCI is capable of delivering large amounts of coolant to refill the vessel once the system depressurizes. The LPCI consists of three residual heat removal pumps, each of which injects coolant through separate piping into the recirculation loops.

The ADS activates about one-third of the safety relief valves in a BWR/4. These valves are opened to reduce the vessel pressure to mitigate the consequences of small breaks where depressurization is required to actuate the LPCI and LPCS.

D.3.2. Scenario Description

As described in Ref. D-7, a LOCA in a BWR is defined as an instantaneous break in the system with break sizes up to and including a double-ended severance of the recirculation loop piping. Recirculation line breaks produce the highest peak cladding temperatures in BWRs. As such, a double-ended guillotine break in the recirculation line for a BWR/4 with the unavailability of off-site power is chosen for the discussion below. LPCS, HPCI and LPCI are credited in the simulation.

As described in Ref. D-7, off-site power is assumed to be lost at the time of the break opening. Following reactor trip the core power decreases to the fission product decay heat.

Following opening of the break, the vessel pressure and core flow initially decrease. Because the energy expelled out the break and through the steam lines exceeds the energy deposited into the coolant from the core, the system depressurizes over the first few seconds. Very little mass is assumed to enter the system during this period because the feedwater flow is assumed to coast to a zero value in one second. At about 5 seconds, the main steam isolation valves are assumed to be completely closed, preventing steam from exiting the vessel. The closure of the main steam isolation valves causes the partial repressurization and the elevated system pressures during the first 10 seconds of the event.

The initial rapid loss in core flow is due to the opening of the break in the recirculation loop. However, the intact loop pump does coast down during the event and influences the core flow behavior during the early portion of the transient. When the suction to the jet pumps uncover, the core flow rapidly decreases. And, upon uncovery of the suction nozzle to the recirculation line, the volumetric flow rate through the break in this location increases significantly, causing an increase in the system depressurization rate. This increased depressurization after about 10 seconds causes the subcooled liquid in the lower plenum to eventually saturate and flash. Figure D-7 presents the lower plenum liquid mass and the decrease in inventory upon flashing at about 11 seconds into the transient. The flashing of the fluid in the lower plenum causes an associated increase in the core flow at about 11 seconds. The jet pump discharge mass flow rates display the early flow reversal on the broken loop side after the break opens. The downcomer liquid level rapidly decreases due to the opening of the break and the effect of lower plenum flashing at 11 seconds.

The break mass flow rate decreases as the suction line uncovers early in the event.

The clad temperature responses for the low, average and high power rods are given in Fig. D-8. The clad temperature is governed by the core flow early in the event as nucleate boiling governs the heat removal from the fuel rods during the initial portion of the event. As the core flow achieves a low flow condition, boiling transition develops as the core flow degrades and is a direct result of the uncovery of the jet pump discussed above. The heat transfer reaches film boiling, and with uncovery of the hot spot at about 25 seconds, the clad temperature for the high-powered rod begins to increase due to the low heat-transfer coefficients characteristic of steam cooling. The cladding temperature continues to increase at a rapid rate until the ECCS initiates injection into the reactor vessel initiating refill at about 46 seconds as noted in Fig. D-7.

Coolant enters the core peripheral bundles from the low-pressure core spray that condenses steam and pools in the upper plenum. The downflow of ECC injection (countercurrent flow) through the outer lower-power bundles initiates refill of the lower plenum. Because of the high steaming rates in the hotter fuel bundles, downflow

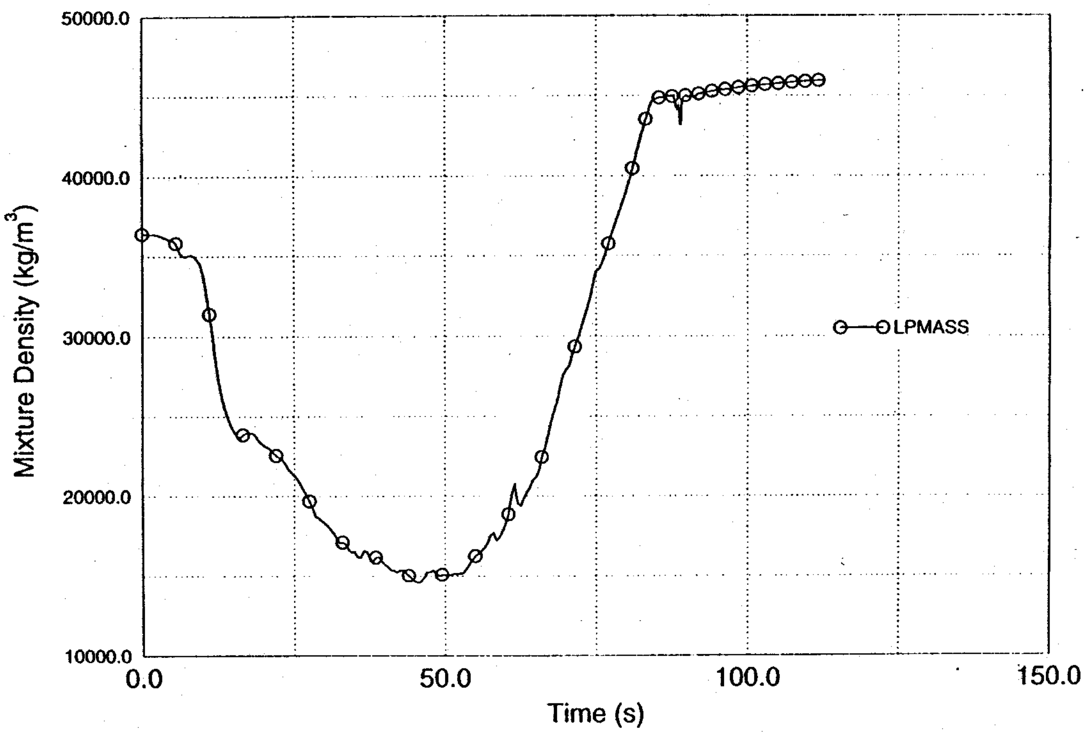


Fig. D-7. Lower-plenum fluid mass.

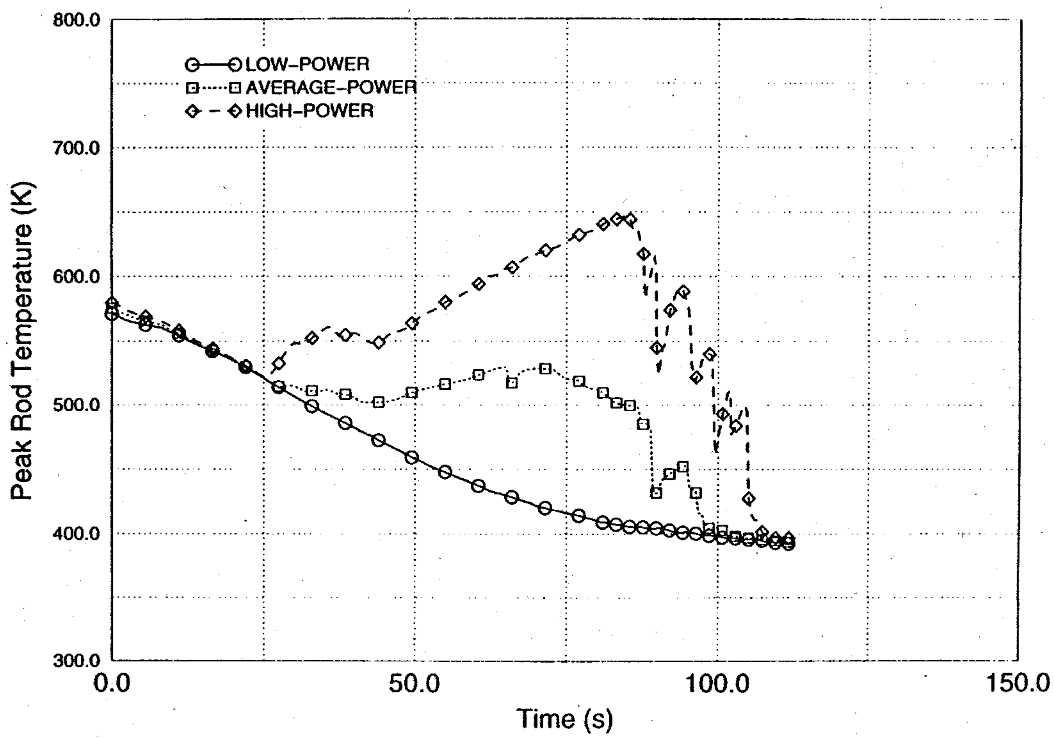


Fig. D-8. Peak cladding temperatures.

of liquid is precluded in the central core region. That is, the high steam velocities in these regions preclude counter-current flow through the upper tie plate and the channel inlet orifices. Once the lower plenum refills and flashing in this region subsides, reflood of the core central bundles begins at about 80 seconds in the core. However, until sufficient coolant enters the core, heat removal from the bundles in the interior of the core is controlled by forced convection to steam and thermal radiation. As sufficient coolant enters the core interior hot bundles, the droplets entrained in the steam eventually provide sufficient cooling to terminate the cladding heat-up as dispersed flow film boiling governs the heat removal from the upper portion of the fuel rods. As the coolant injection into the core continues during this reflood period, the core eventually quenches and the heat transfer returns to nucleate boiling, where the clad temperatures remain within several degrees of the coolant saturation temperature during the long term. Once sufficient coolant has entered the core's high-power region, the peak clad temperature is terminated and quench occurs at 107 seconds, as noted in Fig. D-8.

Early in the event, the two-phase level in the vessel remains at elevated values due to the early depressurization and attendant flashing of the liquid in the core. Following uncovery of the jet pump and the later lower-plenum flashing, the fluid lost through the break, along with the flashing and boiling in the core region, causes the upper portions of the fuel bundles to uncover. Following lower-plenum flashing and the continued depressurization of the system, the ECC is activated and coolant begins to enter and refill the vessel. Refill is initiated by the liquid downflow through the low-powered peripheral. The low- and average-powered core region bypass regions display this similar downward flow behavior. Reflood of the core begins after refill of the lower plenum and the clad temperature excursion is finally terminated at about 85 seconds into the event. Once sufficient coolant has entered the fuel bundles, fuel rod quenching is initiated. The heat transfer returns to nucleate boiling, which maintains the core in a cooled condition for the long term.

REFERENCES

D-1. B. E. Boyack, "AP600 Large-Break Loss-of-Coolant Accident Phenomena Identification and Ranking Tabulation," Los Alamos National Laboratory document LA-UR-95-2718 (August 1995).

D-2. J. F. Lime and B. E. Boyack, "Updated TRAC Analysis of an 80% Double-Ended Cold-Leg Break for the AP600 Design," Los Alamos National Laboratory LA-UR-95-1594, (June 1995).

D-3. "NRC Plant Information Book," U. S. Nuclear Regulatory Commission (January 2000).

D-4. S. M. Bajorek, A. Ginsberg, D. J. Shimeck, K. Ohkawa, M. Y. Young, L. E. Hochreiter, P. Griffith, Y. Hassan, T. Fernandez, and D. Speyer, "Small Break Loss Of Coolant Accident Phenomena Identification and Ranking Table (PIRT) for Westinghouse Pressurized Water Reactors," Proceedings of the 1999 NURETH-9 Conference, San Francisco, California (October 3-8, 1999).

D-5. M. G. Ortiz and L. S. Ghan, "Uncertainty Analysis of Minimum Vessel Liquid Inventory during a Small-Break LOCA in a B&W Plant – An Application of the CSAU Methodology Using the RELAP5/MOD3 Computer Code," EG&G Idaho, Inc. document NUREG/CR-5818 (December 1992).

D-6. M. G. Ortiz, L. S. Ghan, and G. E. Wilson, "Development of a Phenomena Identification and Ranking Table (PIRT) of Thermal-Hydraulic Phenomena During a PWR Small Break LOCA," EG&G Idaho, Inc. document EGG-EAST-9080 (May 1990).

D-7. M. Straka and L. W. Ward, "BWR PIRT and Assessment Matrices for BWR LOCA and Non-LOCA Events," Scientech, Inc. document SCIE-NRC-393-99 (1999).

APPENDIX E

OTHER STANDARD TEST PROBLEM SPECIFICATION EXAMPLES

Test problems developed by J. Mahaffy of Pennsylvania State University are summarized in this appendix. These problems illustrate several aspects of validation using standards other than those that employ experimental data as discussed in Sections 2 and 6 of the main report.

E.1. Static Vessel Test Problem

Purpose: The purpose of this problem is to test for anomalies in the 3D momentum transport terms that can result in spurious circulation patterns. It is an important test to qualify the code for use on passive reactors.

Success Metric: Fluid velocities should be observed at all positions in the final large edit and compared with the expected zero flow. In addition, the void fraction in level 11 should be compared with the expected value of 0.50.

Problem Description: This problem consists of a PWR vessel connected via short, single-cell pipes to zero flow boundary conditions on the cold legs, and constant atmospheric pressure conditions on the hot legs. All temperatures in the system are set to 300 K. All pressures are initialized to 0.1 MPa. The hot- and cold-leg pipes are initially full of air, and the vessel contains water up through the midpoint of the second level below the cold (or hot) legs. The upper vessel is filled with air.

Under ideal conditions, this problem undergoes a brief transient to adjust the pressures to appropriate hydrostatic values and then settles into a steady configuration with no flow and level water surface.

The vessel model used was obtained from the USPWR test problem. It has 4 radial zones with boundaries at 1.0919, 1.6855, 1.9376, and 2.1971 meters and 8 evenly spaced azimuth zones. There are 17 axial zones with upper faces at the following meters:

1.3672	1.9389	2.4469	3.0882	3.6400	4.2800
5.0100	5.9200	6.7458	7.1395	7.5523	7.9650
8.7015	9.2655	1.0137	10.926	12.575	

The void fraction in the vessel is set to 0 for levels 1 through 10; to 0.5 for level 11; and to 1.0 for levels 12 through 17. The air partial pressure is initially set equal to the total pressure (0.1 MPa) in all cells.

Connections to the vessel are through simple, single-cell PIPE components representing the nozzle sections of the hot and cold legs. The hot legs have a cell length of 0.825 m, cell volume of 0.408 m^3, cell face adjacent to the vessel with 0.8-m^2 area, and cell face adjacent to the BREAK of 0.427 m^2. The cold legs have a cell length of 1.163 m, a cell volume of 0.445 m^3, a cell face adjacent to the vessel of 0.6297 m^2, and a cell face adjacent to the FILL of 0.383 m^2.

The type 3 FILLs connected to the cold-leg PIPEs have geometries (DX and VOL) matching the adjacent PIPE cell. Both inlet velocities are set to 0. The type 0 BREAKs are also given geometries matching the adjacent PIPE cell.

E.2. Bubble Rise Test Problems

Purpose: The primary purpose of these problems is to check the logic for an initial increase above zero void fraction when the gas entering a cell is primarily non-condensable. As the run approaches steady state and timestep size increases, the problem also provides a valuable test of stability associated with interfacial drag.

Additionally, these tests provide a simple assessment of the low-void interfacial drag and can be used as a check for changes in the low-void evaporation model or in numerical diffusion.

Success Metrics: The output of Control block -100 at the end of the second timestep should be compared with the total mass of air that has flowed in from the FILL during the first two timesteps. This is a good flag for problems in the transition from single-phase liquid to bubbly flow.

Time history plots should be compared for timestep size, as should those of the void fraction, air partial, and vapor velocity midway up the tank. This information will permit detection of instabilities. When stable results are obtained, the information can be compared with data on bubble rise velocities and can be used to indicate changes in the evaporation model at low void.

Problem Description: These problems follow bubbles injected into the bottom of a tank of water. The tank is 2 m high and 2 m in diameter and is initially full of water. At time zero, air bubbles are injected at the bottom at velocity of 0.132 m/s from a source that has a void fraction of 0.03. Only the air enters from this source. The air-bubble velocity has been set to match the bubble rise velocity obtained from the present 1D interfacial drag correlation in TRAC for a void fraction of 0.03. It should be changed if the interfacial drag correlation is changed. The liquid velocity at this boundary is set to zero. The top of the tank is bounded by a pressure boundary condition of 0.1001 MPa. All temperatures in the tank and boundary conditions are 300 K. After ~15 s, the bubbles have spread uniformly through the tank, and a steady state should follow.

In all decks, a type 3 FILL supplies the air. The FILL's total pressure and air partial pressure are set to 0.1001 MPa. Its liquid and vapor temperatures are set to 300 K, and its void fraction set to 0.03. The volume of the fill is 0.3145926 m^3, and the length is 0.1 m. The liquid fill velocity is zero and the vapor velocity is 0.132 m/s. All decks also share the same upper-boundary pressure condition. This is provided with a type 0 BREAK, which has the same pressure, temperatures, void fraction, and geometry as the FILL.

The 1D versions of the test problems model the tank with a 20-cell PIPE. Each cell is 0.1 m long, with cell volume and cell edge flow areas calculated automatically from the 2-m, hydraulic diameter and the assumption of a uniform, circular cross section (FA and VOL set to -1.0 in the input). Two 1D problems have been created that differ only in the final maximum timestep size. One input deck runs to steady state, while the second develops a bounded instability due to a higher requested maximum timestep. As the code is improved, the final maximum timestep should be increased to maintain one stable problem and one with instability.

The 3D versions of the problem replace the central 18 cells of the pipe with an equivalent vessel configured with 1 radial ring, 1 theta zone, and 18 axial levels, each 0.1 m high. One pair of problems results in stable and unstable runs analogous to those for the pure 1D. A final problem was created with the new reflood model activated (the rod temperatures are all 300 K).

E.3. Falling Droplet Test Problems

Purpose: The two primary purposes of this test series are to check logic for initial decrease from a void fraction of one- to two-phase dispersed flow, and to test for stability problems associated with interfacial drag. Additionally, this test provides a simple assessment of the low-void interfacial drag and can be used as a check for changes in the low-void evaporation model or in numerical diffusion.

Success Metrics: Computed total system mass should be compared for all large edits in the calculation, with particular attention paid to the first three edits. This is used as a flag for problems in the transition from single-phase gas to dispersed flow.

Time history plots should be compared for timestep size, as should those of the void fraction, air partial and vapor velocity midway up the standpipe. This information will permit detection of instabilities. When stable results are obtained, the information can be compared with data droplet velocities and can be used to indicate changes in the evaporation model at high void.

Problem Description. This problem follows drops injected at the top of an air-filled standpipe. The standpipe is 2 m high and 2 m in diameter and initially contains only air. At time zero, water is injected into the top of the pipe at velocity of 0.2287 m/s from a source that has a void fraction of 0.99. Only the liquid enters from this source. This velocity has been set to match the droplet terminal velocity obtained from the current 1D interfacial drag correlation in TRAC for a void fraction of 0.99. This injection velocity should be changed if the interfacial drag correlation is changed. The gas velocity at this upper boundary is set to zero. The bottom of the standpipe is connected to a pressure boundary condition of 0.100 MPa. All temperatures in the standpipe and boundary conditions are 300 K. After ~10 s, the droplets have spread uniformly through the system and a steady state should follow.

In all decks a type 3 FILL supplies the liquid. The FILL's total pressure and air partial pressure are set to 0.100 MPa. Its liquid and vapor temperatures are set to 300 K, and its void fraction set to 0.03. The volume of the fill is 0.3145926 m^3, and the length 0.1 m. The gas fill velocity is 0, and the liquid velocity is 0.2287 m/s. All decks also share the same lower-boundary pressure condition. This is provided with a type 0 BREAK, which has the same pressure, temperatures, void fraction, and geometry as the FILL.

The 1D version of these test problems models the standpipe with a 20-cell PIPE. Each cell is 0.1 m long, with cell volume and cell edge flow areas calculated automatically from the 2-m, hydraulic diameter and the assumption of a uniform circular cross section (FA and VOL set to -1.0 in the input). Only a single 1D problem has been created because the code is stable for all timestep sizes currently permitted. As the code timestep control is improved, a second 1D test may be needed to mark a threshold of instability.

The 3D versions of the problem replace the central 18 cells of the pipe with an equivalent vessel configured with 1 radial ring, 1 theta zone, and 18 axial levels, each 0.1 m high. One pair of problems results in stable and unstable runs analogous to those for the pure 1D. A final problem pair was created with the new reflood model activated (although the rod temperature is 300 K).

E.4. Boron Transport

Purpose: The primary purpose of this test set is to provide a quantitative measure of the numerical diffusion associated with the code's boron transport equations. It has as a secondary purpose the introduction of a method by which the numerical diffusion of any of the mass or energy equations may be measured.

Success Metrics: The key output variable is the value of control block -120. The numerical value should be that predicted by the C-curve method for the conditions used in the calculation. The final value is of prime interest; however, a time history plot of this variable should be examined to be certain that it has ceased to change.

Problem Description: This problem models the propagation of a 1-s-long square pulse of boron with a peak concentration of 0.002 and a base concentration of 0. Flow is through a pipe 10 m in length and 1 m in diameter. Velocity of the pure liquid flow is maintained at 2.0 m/s. Temperature of the liquid is 577.6 K, and pressure at the outlet is fixed at 15.51 MPa.

A type 10 FILL drives flow. Input is set to only take boron concentration from a control block, other variables are taken as constants. The FILL void fraction is fixed at 0, the liquid velocity is at 2.0 m/s, the liquid temperature is at 577.6 K, and pressure is at 15.51 MPa. The volume associated with the FILL is 0.785398 m^3, and the length is 1.0 m. The control block supplying boron concentration (CB -5) is simply a table with entries of 0.002 at 0 and 1 s and 0 at 1.001 and 10000.0 s.

The PIPE component has 20 cells each 0.5 m long with cell volumes and face areas computed internally from the 1.0-m hydraulic diameter. Initial conditions in the pipe are set to give velocity of 2 m/s at all faces, and temperature of 577.6 K, pressure of 15.51 MPa, and void fraction of 0 in all cells.

Conditions at the PIPE outlet are provided by a type 0 BREAK component. Fluid conditions and geometry of the BREAK match those of the FILL, except that boron concentration is fixed at 0.

A key feature of the test problem is a set of control blocks (-1, -2, and -10 through -120) that implement the C-curve method to provide a quantitative measure of the numerical diffusion associated with the propagation of the boron. The method was originally developed for analysis of experimental data on turbulent mixing (Levenspiel, "Chemical Reaction Engineering," Second Edition, Wiley, 1972) and has been adapted for quantifying numerical diffusion.

APPENDIX F

CANDIDATE TESTS FOR THE TRAC-M COMMON LBLOCA VALIDATION TEST MATRIX

In this appendix, we present the candidate experimental facilities for the TRAC-M common LBLOCA validation test matrix. For each PIRT local-level (LL) process/phenomena identified in Section 4 (Table 4-1), we provide a table. Each table lists the experimental facilities that have produced data, which are candidates for inclusion in the validation test matrix. Where possible, specific tests have been identified, but we acknowledge that more effort is required in this regard. Local-level PIRT phenomena are covered in Tables F-1 through F-15. Component- and system-level PIRT phenomena are covered in Tables F-16 through F-22.

TABLE F-1
CANDIDATE COMMON EXPERIMENTAL FACILITIES: BOILING–FILM

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Film				
	Plant Range	Test Facility			
Plant Parameter		Stewart	Laperriere	Winfrith	THEF/INEL
P (MPa)	0.2–15.4	0.009 - 2.03	9.6 - 10	0.2 - 7	0.2 - 7
q (W/cm^2)	1–46	0.16 - 0.19	0.17 - 0.4	1 - 30	0.8 - 22.5
v (m/s)	0–4				
G (kg/m^2-s)	10–2455	360 - 2783	2815 - 4406	50 - 2000	12 - 70
Comments		Ref. 2: Fundamental tube data	Ref. 3: Fundamental tube data	Fundamental tube data. Ref. 4 facility #10.4	Fundamental tube data. Ref. 4 facility #11.3

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Film				
	Plant Range	Test Facility			
Plant Parameter		Fung/U. of Ottawa	Lehigh	TPTF/JAERI	Blowdown HT/RS37
P (MPa)	0.2–15.4	0.1	0.1 - 1.0	0.5 - 12	1.3 - 15
q (W/cm^2)	1–46		< 10	3 - 25	74 - 163
v (m/s)	0–4				
G (kg/m^2-s)	10–2455		< 300	20 - 410	3300 - 3828
Comments		Ref. 5: Fundamental tube data, includes void fraction.	Fundamental rod-bundle data. Ref. 4 facility #11.42.	BWR and PWR core geometries; Ref. 4 facility #6.1.	25-rod bundle; Ref. 4 facility # 4.5.

Nomenclature
P, pressure

q, heat flux
v, velocity
G, mass flux

<u>References</u>

1. B. E. Boyack, “TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models,” Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
2. J. C. Stewart, “Low Quality Film Boiling at Intermediate and Elevated Pressures,” M.Sc. thesis, University of Ottawa, Ottawa, Canada (1981).
3. A. Laperriere, “An Analytical and Experimental Investigation of Forced Convective Film Boiling,” M.Sc thesis, University of Ottawa, Ottawa (1983).
4. “Separate Effects Test Matrix for Thermal-Hydraulic Code Validation,” Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
5. K. K. Fung, “Subcooled and Low Quality Film Boiling of Water in Vertical Flow at Atmospheric Pressure,” Ph.D. Thesis, University of Ottawa (1981).

TABLE F-2
CANDIDATE COMMON EXPERIMENTAL FACILITIES: BOILING–TRANSITION

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Transition (Note 1)				
	Plant Range	Test Facility			
Plant Parameter		U. of Cincinnati	Argonne SGTF	U. of Ottawa	Johannsen
P (MPa)	0.2–15.4	0.1 - 0.4	7 - 15.3	0.1	0.1 - 1.2
q (W/cm^2)	1–46	2 - 75		40 - 250	20 - 800
v (m/s)	0–4				
G (kg/m^2-s)	10–2455	7.3 - 144	70 - 320	68 - 203	25 - 200
Comments		Refs. 2-3: Fundamental tube and annulus data (Note 3)	Ref. 4: Fundamental tube data (Note 3)	Ref. 5: Fundamental tube data	Ref. 7: Fundamental tube data

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Transition (Note 1)				
	Plant Range	Test Facility			
Plant Parameter		Bennett	FZK Single Rod	NEPTUN	
P (MPa)	0.2–15.4	6.9	0.1	0.41	
q (W/cm^2)	1–46	7 - 100	0–56		
v (m/s)	0–4				
G (kg/m^2-s)	10–2455	< 5500	150	15–150	
Comments		Ref. 8: fundamental tube data	Refs. 9-10 Note 4	Refs 9, 11: rod bundle tests 5036 and 5050, Note 4	

Nomenclature
P, pressure
q, heat flux
v, velocity
G, mass flux

References

1. K. Johannsen, “Low Quality Transition and Inverted Annular Flow Film Boiling of Water: An Updated Review,” Experimental Thermal and Fluid Science, Vol. 4, pp. 497-509 (1991).
2. S. Wang, “A Study of Transition Boiling Phenomnena with Saturated Water at 1-4 Bar,” Ph.D Thesis, College of Engineering, University of Cincinnati, Ohio (1981).
3. S. Wang, Y. K. Kao, J. Weisman, “Studies of Transition Boiling Heat Transfer in a Vertical Round Tube,” Nuclear Engineering Design, Vol. 70, pp. 223-243 (1982).
4. D. M. France, I S. Chan, and S. K. Shin, “High-Pressure Transition Boiling in Internal Flows,” J. Heat Transfer, Vol. 109, pp. 498-502 (1987).
5. S. C. Cheng, W. W. L. Ng, K. T. Heng, and D. C. Groeneveld, “Measurements of Transition Boiling Data for Water Under Forced Convective Conditions,” Transactions of the ASME, Journal of Heat Transfer, Vol. 100, pp. 382-384 (May 1978).
6. “Separate Effects Test Matrix for Thermal-Hydraulic Code Validation,” Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
7. K. Johannsen, P. Weber, and Q. Feng, “Experimental Investigation of Heat Transfer in the Transition Boiling Region,” Technische Universitat Berlin document EUR-13135 (October 1990).
8. A. W. Bennett, G. F. Hewitt, H. A. Kearsey, and R. K. F. Keeys, “Heat Transfer to Steam-Water Mixtures Flow in Uniformly Heated Tubes in Which the Critical Heat Flux Has Been Exceeded,” Atomic Energy Research Establishment document AERE-R-5373 (March 1968).
9. E. Elias, V. Sanchez, and W. Hering, “Development and Validation of a Transition Boiling Model for RELAP5/MOD3 Reflood Simulation,” Nuclear Engineering and Design, Vol. 183, pp. 269-286 (1998).
10. P. Hoffman and V. Noack, “Experiment on the Quench Behavior of the Fuel Rod Segments,” Second International Quench Workshop, Karlsruhe (September 1996).
11. M. Richner, G. Th. Analytis, and S. N. Aksan, “Assessment of RELAP5/MOD2, cycle 36.02, Using NEPTUN Reflooding Experimental Data,” Paul Scherrer Institut document PSI104, UREG/IA-00103 (October 1991).

Notes

1. In his review (Ref. 1), Johannsen states “The main conclusions of Refs. 1-5: There is a lack of a reliable empirical database for heat transfer in the transition and inverted annular flow film boiling region, especially at low flows and pressures; the available correlations and analytical models are not very accurate; and problems still exist in understanding the physical mechanisms.”

2. The OECD/CSNI separate effect test matrix report (Ref. 6) identifies tests for “Heat Transfer: Post-CHF in the Core . . “ but does not subdivide the post-CHF area further to identify tests that may have usable data for validating the transition boiling model.
3. Per Ref. 4: “It is important to differentiate between transition boiling phenomena in internal and external flows where the hydrodynamics are significantly different.”
4. Used for validation of RELAP5/MOD3 transition boiling model (See Ref. 9). Data for NEPTUN Test 5050 is in the NEA data bank.

TABLE F-3
CANDIDATE COMMON EXPERIMENTAL FACILITIES: CONDENSATION–INTERFACIAL

Plant	Westinghouse 4-Loop PWR
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Refill
PIRT Parameter	Condensation–Interfacial heat and mass transfer

	Plant Range	Test Facility			
Plant Parameter		Lee, et al.	Kim, et al.	Akimoto, et al.	Celata, et al.
P (MPa)	0.1	1.0		0.05–0.2	0.1–1.0
G_g, (kg / m^2-s)			Re_v =2500–30000	0–74	to 20 kg / hr
G_l, (kg / m^2-s)			Re_l =800–15000	0–1000	To 120 kg / hr
Superheat (K)		10–40			40
Comments		Ref. 1: Cocurrent stratified horizontal condensing flows (See Note 1)	Ref. 2: counter-current steam-water stratified flow (See Note 2)	Ref. 3-4: water injected into flowing steam at 90° angle.	Ref. 5-6: near stagnant superheated steam condensing on a slowly-moving subcooled water surface

Nomenclature
P, pressure
G_g, gas mass flux
G_l, liquid mass flux

References

1. L. Lee, R. Jensen, S. G. Bankoff, M. C. Yuen, and R.S. Tankin, Local Condensation in Cocurrent Steam-Water Flow," *Nonequilibrium Interfacial Transport Processes* (edited by J. C. Chen and S.G. Bankoff) ASME, New York (1979).
2. H. J. Kim, S. C. Lee, and S. G. Bankoff, "Heat Transfer and Interfacial Drag in Countercurrent Steam-Water Stratified Flow," International Journal of Multiphase Flow, Vol. 11, pp. 593-606 (1985).
3. H. Akimoto, Y. Tanaka, Y. Kozawa, A. Inoue, and S. Aoki, "Oscillatory Flows Induced by Direct Contact Condensation of Flow Steam with Injected Water," Journal of Nuclear Science and Technology, Vol. 22, No. 4, pp. 269-283 (April 1985).
4. H. Akomoto, T. Kozwa, A.Inoue, and S. Aoki, "Analysis of Direct-Contact Condensation of Flow Steam onto Injected Water with Multifluid Model of Two-Phase Flow," Journal of Nuclear Science and Technology, Vol. 20, No. 12, pp. 1006-1022 (1983).

5. G. P. Celata, M. Cumo, G. E. Farello an G. Focardi, "Direct Contact Condensation of Superheated Steam on Water," International Journal of Heat and Mass Transfer, Vol. 30, No. 3, pp. 449-458 (1987).
6. G. P. Celata, M. Cumo, G. E. Farello an G. Focardi, "A Theoretical Model of Direct Contact Condensation on a Horizontal Surface," International Journal of Heat and Mass Transfer, Vol. 30, No. 3, pp. 459-467 (1987).

Notes
1. Inlet liquid temperatures were between 30 and 62˚C.
2. Conducted at aspect ratios between 4 and 87 degrees. Vapor and liquid Reynolds numbers reported as between 2,500–30,000 and 800–15,000, respectively.

TABLE F-4
CANDIDATE COMMON EXPERIMENTAL FACILITIES: DRAINING

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Long-Term Cooling				
PIRT Parameter	Draining				
	Plant Range	Test Facility			
Plant Parameter		Foster	Lubin and Springer	Georgia Institute of Technology	
P (MPa)	0.2	0.1	0.1		
q (W/cm^2)					
G (kg/m^2-s)	0.0–4150 (note 1)	0.0–4150 (specify)	1580		
Comments		Ref. 1: formula provides the time to empty a vertical cylinder, the top of which is open to atmosphere	Ref. 2: SET experiment-data on draining water from a 5-1/2 in cylinder, the top of which is open to atmosphere	Refs. 3-4: SET experiment for draining of a sealed vertical cylinder induces 2-phase countercurrent flow	

	Plant Range	Test Facility	
Plant Parameter		ROSA-AP600	PACTEL
P (MPa)	0.2	0.2–7	
q (W/cm^2)			
G (kg/m^2-s)	0.0–4150 (note 1)		
Comments		Ref. 5: IET experiments (note 2). Need to acquire actual data reports.	Ref. 6: IET experiments in PACTEL, a scaled IET of a 6-loop VVER-440 type PWR. Assessment would also demonstrate adequacy of TRAC for this plant application

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. T. C. Foster, "Time Required to Empgy a Vessel," Chemical Engineering, Vol. 95, No. 5, pp. 171-172 (1990).
2. B. T. Lubin and G. S. Springer, "The Formation of a Dip on the Surface of a Liquid Draining From a Tank," Journal of Fluid Mechanics, Vol. 29, Part 2, pp. 385-390 (1967).
3. K. H. Lillibridge, S. M. Ghiaasiaan, and S. I. Abdel-Khalik, "An Experimental Study of Gravity-Driven Countercurrent Two-Phase Flow in Horizontal and Inclined Channels," Nuclear Technology, Vol. 105, pp. 123 (1994).
4. S. M. Ghiaasiaan, B. K. Kamboj, and S. I. Abdel-Khalik, "Modeling of Gravity-Driven Oscillatory Countercurrent Two-Phase Flows," Nuclear Science and Technology, Vol. 117, pp. 22-32 (1994).
5. T. Yonomoto, M. Kondo, Y. Kukita, L. S. Ghan, and R. Schultz, "Core Makeup Tank Behavior Observed During the ROSA-AP600 Experiments," Nuclear Technology, Vol. 119, pp. 112-122 (August 1997).
6. J. Tuunanen, V. Riikonen, J. Kouhia, and J. Vihavainen, "Analysis of PACTEL Passive Safety Injection Experiments GDE-21 through GDE-25," Nuclear Engineering and Design, Vol. 180, pp. 67-91 (1998).

Notes

1. Based upon TRAC-PF1/MOD2 intermediate break loss-of-coolant accident (LA-UR-95-1785). Maximum IRWST flows are 100 kg/s and 30 kg/s for the broken and intact loops, respectively. Maximum broken loop CMT flow is 50 kg/s. IRWST delivery line is 0.15405-m diameter. CMT delivery line is 0.17305-m diameter.
2. Reference 5 lists the following experiments as demonstrating a variety of Core Makeup Tank processes (SB1, CL4, CL3, CL6, CL7, CL5, PB2, SG1, DV1, CL8, PB1, AD1, and SG2).

TABLE F-5
CANDIDATE COMMON EXPERIMENTAL FACILITIES: ENTRAINMENT/DEENTRAINMENT

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, Reflood				
PIRT Parameter	Entrainment / deentrainment				
	Plant Range	Test Facility			
Plant Parameter		Cousins and Hewitt	Steen and Wallis	Lopez de Bertodano et al.	Paras and Karabelas
P (MPa)	0.2	0.22		0.14–0.66	
j_f (m / s)		0.06–0.39	0.08–0.319	0.074–0.54	0.02–0.2
j_g (m / s)		24-47		24.5–126	31–66
Comments		Ref. 1, 3: upward flow air-water in vertical round tube	Ref. 2, 3: downward air-water flow in 1.07 to 1.59-cm tubes	Ref. 4-5: adiabatic upward flow air-water loop.	Ref. 6: adiabatic horizontal air-water flow

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, Reflood				
PIRT Parameter	Entrainment / deentrainment				
	Plant Range	Test Facility			
Plant Parameter		Williams			
P (MPa)	0.2				
j_f (m / s)					
j_g (m / s)					
Comments		Ref. 7: adiabatic horizontal air-water flow in pipe			

Nomenclature
P, pressure
j_f, liquid phase volumetric flux (superficial velocity)
j_g, gas phase volumetric flux (superficial velocity)

References

1. L. B. Cousins and G. F. Hewitt, "Liquid Phase Mass Transfer in Annular Two-Phase Flow: Droplet Deposition and Liquid Entrainment," United Kingdom Atomic Energy Authority Report AERE-R5657 (1968).
2. D. A. Steen and G. B. Wallis, "The Transition from Annular to Annular-Mist Concurrent Two-Phase Down Flow," Atomic Energy Commission Report NYO-3114-2 (1964).
3. M. Ishii and K. Mishima, "Droplet Entrainment Correlation in Annular Two-Phase Flow," International Journal of Heat and Mass Transfer, Vol. 32, No. 10, pp. 1835-1846 (1989).
4. M. A. Lopez de Bertodano, C.-S. Jan, and S. G. Beus, "Annular Flow Entrainment Rate Experiment in a Small Vertical Pipe," Nuclear Engineering and Design, Vol. 178, pp. 61-70 (1997).
5. A. Assad, C. Jan, M. Lopez de Bertodano, and S. Beus,"Scaled Entrainment Measurements in Ripple-Annular Flow in a Small Tube," Nuclear Engineering and Design, Vol. 184, pp. 437-447 (1998).
6. S. V. Paras and A. J. Karabelas, "Droplet Entrainment and Deposition in Horizontal Annular Flow," International Journal of Multiphase Flow, Vol. 17, No. 4, pp. 455-468 (1991).
7. L. R. Williams, "Entrainment Measurements in a 4-Inch Horizontal Pipe," University of Illinois M.Sc. Thesis (1986).

TABLE F-6
CANDIDATE COMMON EXPERIMENTAL FACILITIES: EVAPORATION

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Blowdown, Refill, Reflood
PIRT Parameter	Evaporation

	Plant Range	Test Facility			
Plant Parameter			Allesandrini, et al.	Wurtz	Hewitt
P (MPa)	0.2–15.4		5.0	7.0	Low pressure
q (W/cm^2)	1- 46		Adiabatic	Adiabatic	61-65
G (kg/m^2-s)	0–2455		1500	500-1000	297
Subcooling (K)					
Comments			Ref. 2: See Note 1 Steam-water data	Ref. 3: See Note 1 Steam-water data	Ref. 4: See Note 1 non-equilibrium entrainment data

	Plant Range	Test Facility			
Plant Parameter		Becker	Lehigh Tube	INEL	Winfrith
P (MPa)	0.2–15.4	1-16		0.48 – 7.07	0.199 – 1.009
q (W/cm^2)	1- 46	10-300		0.8 – 22.5	1 - 30
G (kg/m^2-s)	0–2455	500-3000		12.1 - 70.7	51 - 2014
Subcooling (K)		10			
Comments		Ref. 5: See Note 2 Single tube-diameter and length 0.015 and 7 m, respectively; 5 different heat flux profiles.	Refs. 6-7: Internal flow in heated tube using hot-patch technique.	Refs. 8-9: Internal flow in heated tube using hot-patch technique Also entered for film boiling.	Refs. 10-11: Internal flow in heated tubes. Also entered for film boiling.

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Evaporation				
	Plant Range	Test Facility			
Plant Parameter		Lehigh Bundle	Flecht-Seaset		
P (MPa)	0.2–15.4	0.105 – 0.120			
q (W/cm^2)	1- 46	< 10			
G (kg/m^2-s)	0–2455	< 300			
Subcooling (K)		0.4 - 40			
Comments		Ref. 12: 3x3 rod bundle with 98 fixed-CHF points and 278 slow-moving CHF data points. Wall temperatures and heat fluxes vs distance above the quench front. Vapor superheats at two axial locations. Used hot-patch technique. Also entered for film boiling.	Ref. 13: Use forced-reflood bundle experiment 31504. Flecht-Seaset used a core simulator consisting of 161 electrically heated rods within a 17x17 square matrix.		

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. Removed
2. Alessandrini, G. Peterlongo, and R. Ravetta, "Large Scale Experiments on Heat Transfer and Hydrodynamic with Steam-Water Mixture, Critical Heat Flux and Pressure Drop measurements in Round Vertical Tubes at the Pressure of 51 kg/cm2," Centro Informazioni Studi Experienze report CISE-R 86 (1963).
3. J. Wurtz, "An Experimental and Theoretical Investigation of Annular Steam Water in Tubes and Annuli at 30 to 90 Bar," RISO report 372 (1978).
4. G. F. Hewitt, "Annular Flow Evaporation, Selected Experimental Data Set No. 12," Second International Workshop on Two-Phase Flow Fundamentals Physical Benchmark, Troy, New York (1987).
5. K. M. Becker, P. Askeljung, S. Hedberg, B. Soderquist and U. Kahlbom, "An Experimental Investigation of the Influence of Axial Heat Flux Distributions on Post Dryout Heat Transfer for Flow of Water in Vertical Tubes," Royal Institute of Technology, Department of Nuclear Reactor Engineering Report KTH-NEL-54, presented at the European Two-Phase Flow Group Meeting, Stockholm, June 1-3, 1992.
6. D. G. Evans, S. W. Webb, and J. C. Chen, "Axially Varying Vapor Superheats in Convective Film Boiling," Journal of Heat Transfer, Transactions of the ASME, Vol. 107, pp. 663-669 (1985).
7. D. G. Evans, J. C. Chen, and S. W. Webb, "Measurement of Axially Varying Nonequilibrium in Post-Critical-Heat-Flux Boiling in a Vertical Tube," Vol. 1, NUREG/CR 3363 (1983).
8. R. C. Gottula, K. G. Condie, R. K. Sundaram, S. Neti, J. C. Chen, and R. Nelson, "Forced Convection, Nonequilibrium, Post-CHF Heat Transfer," Transactions of Twelfth Water Reactor Safety Research International Meeting, Gaithersburg, Maryland (1985).
9. R. C. Gottula, K. G. Condie, R. K. Sundaram, S.Neti, J. C. Chen and R. A. Nelson, "Forced Convective, Nonequilibrium, Post-CHF Heat Transfer Experiment Data and Correlation Comparison report," NUREG/CR-3193, also EG&G Idaho, Inc. document EGG-2245 (1985).
10. D. Swinnerton, R.A. Savage, and K. G. Pearson, "Heat Transfer Measurements in Steady-State Post-Dryout at Low Quality and Medium Pressure," AEA Thermal Reactor Services, Physics and thermal Hydraulic Division Report AEA-TRS-1045, Winfrith, United Kingdom Atomic Energy Report AEEW-R 2503 (1990).
11. D. Swinnerton, M. L. Hood, and K. G. Pearson, "Steady State Post-Dryout at Low Quality and Medium Pressure Data Report," Winfrith, United Kingdom Atomic Energy Report AEEW-R 2267 (1988).
12. K. Tuzla, C.Unal, O. Badr, S. Neti, and J. C. Chen, "Thermodynamic Nonequilibrium in post-CHF Boiling in a Rod Bundle," Vols. 1-4, NUREG/CR-4353 (1986).

13. M. J. Loftus, L. E. Hochreiter, C. E. Colnway, C. E. Dodge, A. Tong, E. R. Rosal, M. M. Valkovic, and S. Wong,“PWR FLECHT SEASET Unblocked Bundled, Forced and Gravity Reflood Task Data Report,” U. S. Nuclear Regulatory Commission document NUREG/CR-1532, Electric Power Research Institute document EPRI NP-1459, Westinghouse Electric Corporation document WCAP 9699 (June 1980).

Notes

1. As cited in S. Gao, D. C. Leslie, and G. F. Hewitt, “An Improved TRAC Code for Two-Phase Annular Flow Modeling,” submitted for publication in Nuclear Engineering and Design (1998).
2. As cited in B. J. Azzopardi, “Prediction of Dryout and Post-Dryout Heat Transfer with Axially Non-Uniform Heat Input by Means of an Annular Flow Model,” Nuclear Engineering and Design, Vol. 163, pp. 51-57 (1996).

TABLE F-7
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLASHING–INTERFACIAL

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Blowdown
PIRT Parameter	Flashing–Interfacial heat and mass transfer

	Plant Range	Test Facility			
Plant Parameter		Sozzi and Sutherland	Edwards & O'Brian	CANON SUPER CANON VERTICAL CANON	BNL Nozzle
P (MPa)	5.1–15.4		0.1–7	3.2; 15.0; 13.0	0.7
G (kg/m^2-s)	0–2455				3130–7010
Subcooling (K)					T_{inlet} = 300 K
Comments		Ref. 1: Flashing discharge through a pipe with various entrance characteristics.	Ref. 2: Pipe blowdown	Ref. 3: Pipe blowdown. OECD/SET Facility Numbers 3.3 and 3.4 (See Note 1)	Refs. 4-5: Converging diverging nozzle

	Plant Range	Test Facility			
Plant Parameter		MOBY DICK SUPER MOBY DICK	OMEGA		
P (MPa)	5.1–15.4	0.15–12	0.1–15		
G (kg/m^2-s)	0–2455	4200–10300	W=10-19 kg/s		
Subcooling (K)		Subcooled Saturated	T_{inlet} = 558 K		
Comments		Ref. 3: steady-state critical flow in tubes and nozzles over a spectrum of pressures. OECD/SET Facility Number 3.1, 3.2	Ref. 3: SET test for blowdown of rod bundle. OECD/SET Facility Number 3.15		

Nomenclature
P, pressure
G, mass flux
W, mass flow

References

1. G. L. Sozzi and W. A. Sutherland, "Critical Flow of Saturated and Subcooled Water at High Pressure," General Electric Co. document NEDO-13418 (1975).
2. A. R. Edwards and T. P. O'Brian, "Studies of Phenomena Connected with the Depressurization of Reactors," Journal of the British Nuclear Energy Society, V. 9, pp. 125-135 (1970).
3. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
4. N. Abuaf, B. J. C. Wu, G. A. Zimmer, and P. Saha, "A Study of Nonequilibrium Flashing of Water in a Converging Diverging Nozzle," Vol. 1: Experimental, Vol. 2: Modeling, Brookhaven National Laboratory document NUREG/CR-1864 and BNL-NUREG-51317 (June 1981).
5. P. Saha, N. Abuaf, and B. J. C. Wu, "A Nonequilibrium Vapor Generation Model for Flashing Flows," Transactions of the ASME, Journal of Heat Transfer, Vol. 106, pp. 198-203 (February 1984).

Notes
1. Some of the CANON series of data have been used for TRAC-PF1/MOD1 assessment, and the results are reported in NUREG/IA reports 0001 and 0023.

TABLE F-8
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–CRITICAL

Plant	Westinghouse AP600				
Transient	Large-, Intermediate, and Small-Break Loss-of-Coolant Accident				
Transient Phase	Dependent upon break size				
PIRT Parameter	Critical Flow in Break				
	Plant Range	OECD Test Facility (Ref. 1)			
Plant Parameter	Break	Super Moby Dick (CEA - France)	Rebeca (CEA - France)	Marviken (Sweden)	Piper (University of PISA, DCMN / Italy)
Break Diam. (m)	0.0254 - 0.5588	0.020	0.030	0.2 - 0.509	0.01 - 0.05
Break L/D	1 - >10	0 - 20	0	0.3 to 3.7	
P (MPa)	15.78 - 0.102	2 - 12	0.2 - 0.8	0.1 - 5.2	1 - 9
G (kg/m^2-s)	1.2e06 - 10	8140-62000			
Void fraction	0.0 - 1.0	0 - 0.94	0.981 - 0.999	0 - 1.0	0.0 - 0.9
Subcooling (°C)	71.2 - 0.0	63.8 - 0.0	0	50 - 0	0 - 150
T_{liq} (K)	548.1 - 373.2	421.7 - 597.8			
T_{vap} (K)	619.3 - 400.0	485.5 - 597.8			
	OECD Facility ID	3.2	3.25	8.2	5.17
Comments	Plant parameter ranges are from TRAC AP600 LBLOCA, IBLOCA, and SBLOCA analyses (Refs. 2-4).	Vertical upflow, steady-state facility. Three nozzle configurations tested. Super Moby Dick was one of the critical flow tests used to assess TRAC-PF1/MOD1 Version 14.3 (Ref. 5).	Vertical downflow steady-state facility. Two convergent-divergent nozzle geometries tested. Steam and steam-air mixtures.	Large scale critical flow facility (Ref. 6). A number of nozzle geometries were tested ranging from 0.2 m to 0.509 m in diameter with length-to-diameter ratios from 0.3 to 3.7. TRAC-PF1/MOD2 has been assessed against six tests (Ref. 7).	The Piper facility is primarily for BWR blowdown experiments. The test section is a vertical cylindrical tube, 0.19 m ID, 3m length.

Plant	Westinghouse AP600				
Transient	Large-, Intermediate, and Small-Break Loss-of-Coolant Accident				
Transient Phase	Dependent upon break size				
PIRT Parameter	Critical Flow in Break				
	Plant Range	OECD Test Facility (Ref. 1)			
Plant Parameter	Break	TPFL (Two-Phase Flow Loop, USA)	Critical Flow Rig (GE - USA)	Edwards Blowdown Experiment (UK)	Additional Test Facilities (See Notes).
Break Diam. (m)	0.0254 - 0.5588		0.0127 - 0.0762	0.073	
Break L/D	1 - >10		0.0 - 140.0	56.1	
P (MPa)	15.78 - 0.102	2.0 - 6.0	4.1 - 6.9	6.9 - 0.1	
G (kg/m^2-s)	1.2e06 - 10			17500 - 200	
Void fraction	0.0 - 1.0		0.0 - 0.13	0.0 - 1.0	
Subcooling (°C)	71.2 - 0.0			55.0 - 0.0	
T_{liq} (K)	548.1 - 373.2				
T_{vap} (K)	619.3 - 400.0				
	OECD Facility ID	11.35	11.54		
Comments	Plant parameter ranges are from TRAC AP600 LBLOCA, IBLOCA, and SBLOCA analyses (Refs. 2-4).	Multipurpose support facility to LOFT LOCA experiments. Tee/critical flow experiments performed. The facility has been used for different kinds of experiments but no relevant information is available.	These tests investigated low-quality critical flow, including effects of geometry, length, and L/D. The tests covered 7 different types of nozzles with different nozzle test section lengths. (See Ref. 8)	The Edwards blowdown experiment is not one of the CSNI facilities but is included in the matrix because it simulates a double-ended break of a primary loop pipe. (See Ref. 9)	

Plant	Westinghouse AP600				
Transient	Intermediate, and Small-Break Loss-of-Coolant Accident				
Transient Phase	Dependent upon break size				
PIRT Parameter	Critical Flow in Valves				
	Plant Range	OECD Test Facility (Ref. 1)			
Plant Parameter	ADS Valves	Safety Valve (CISE-SIET, Italy)	Valve Blowdown Facility (CEGB-MEL / UK)	Additional Test Facilities (See Notes).	
Valve Diam. (m)	0.0615 - 0.1767	0.0203, 0.0045			
Valve L/D	>10	0			
P (MPa)	5.5 - 0.102	6.0 - 19.0	28.2		
G (kg/m^2-s)					
Void fraction	0+ - 1.0	0 - 1.0	0 to 1.0		
Subcooling (°C)	0.0				
T_{liq} (K)			513.0		
T_{vap} (K)					
	OECD Facility ID	5.5	10.21		
Comments	Plant parameter ranges are from TRAC AP600 IBLOCA and SBLOCA analyses (Refs. 3-4).	Tested PWR primary loop safety valve behavior in LOCA and operational transients and two-phase flow conditions. Two scaled safety valves tested: (1) 1:7.4 Crosby Type HB valve, 6 M6 orifice and (2) 1:133 SPES pressurizer safety valve.	High flowrate, high pressure test facility for research, development, and testing on primary circuit overpressure protection system valves for the Sizewell B PWR.		

Nomenclature

P Pressure

G Mass Flux

T_{liq} Liquid Temperature

T_{vap} Vapor Temperature

References

1. “Separate Effects Test Matrix for Thermal-Hydraulic Code Validation,” Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).

2. J. F. Lime and B. E. Boyack, “Updated TRAC Analysis of 80% Double-Ended Cold-Leg Break for the AP600,” Los Alamos National Laboratory report LA-UR-95-4431 (January 1996).

3. B. E. Boyack and J. F. Lime, “Analysis of an AP600 Intermediate-Size Loss-of-Coolant Accident,” Los Alamos National Laboratory report LA-UR-95-926 (September 1995).

4. A TRAC AP600 SBLOCA calculation of a 1-in. break in a cold leg was performed in 1996 but the calculation was never published.

5. B. Spindler and M. Pellissier, “Assessment of TRAC-PF1/MOD1 Version 14.3 Using Separate Effects Critical Flow and Blowdown Experiments, Volumes 1 and 2,” USNRC Report NUREG/IA-0023 (SETh/LEML/88-138) (April 1990).

6. R. R. Schultz and L. Ericson, “The Marviken Critical Flow Test Program,” Nuclear Safety, Vol. 22, No. 6, (1981) pp. 712-724.

7. J. L. Steiner and J. F. Lime, “Comparison of TRAC-PF1/MOD2 Calculated Results with Critical-Flow Test Data,” Los Alamos National Laboratory report LA-UR-98-2565 (May 1998).

8. G. L. Sozzi and W. A. Sutherland, “Critical Flow of Saturated and Subcooled Water at High Pressure,” 1975 ASME Winter Annual Meeting Symposium on “Non-Equilibrium Two-Phase Flows” held in Houston, Texas.

9. A. R. Edwards and T. P. O'Brien, "Studies of Phenomena Connected with the Depressurization of Water Reactors, *J. Br. Nucl. Energy Soc.* **9**, 125-135 (1970).

10. E. D. Hughes and B. E. Boyack, “TRAC-P Validiation Test Matrix,” Los Alamos National Laboratory report LA-UR-97-3990 (September 1997).

Notes

There are a number of other facilities selected in the TRAC-P Validation Test Matrix report (Ref. 10) for critical-flow assessment. The following is from the Validation Test Matrix. Not included in the list are those facilities already cited (Super Moby Dick, Marviken, and Critical Flow Rig).

Test Facility	Description
CISE Blowdown	A vertical-pipe blowdown experiment studied depressurization and heat-transfer phenomena of initially flowing subcooled water.
LOFT Valve/Wyle	Studied small-break blowdown from a horizontal round pipe through a 16.0 mm diameter nozzle (may be OECD/CSNI facility 11.5 or 11.34, but no data sheet for either).
ROSA APCL - 03	ROSA 1-inch Cold Leg Break Test.
Carofano-McManus	Studied critical flow of two-phase water at about 0.16 Mpa.
Cumulus Critical Flow	Critical flow of superheated vapor and subcooled liquid through the pressurizer relief valves of a French PWR.
Deich Critical Flow	Studied two-phase critical flow at 0.12 Mpa.
Fincke-Collins Critical Flow	Studied critical flow of subcooled water at pressure from 0.09 to 0.30 MPa.
Neussen Critical Flow	Studied critical flow of two-phase water at pressure from 0.84 to 6.5 Mpa.
VAPORE	Two-phase critical flow through the full-scale automatic depressurization system (ADS) valve trains for the AP600.

TABLE F-9
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–DISCHARGE

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Refill, Reflood
PIRT Parameter	Discharge

	Plant Range	Test Facility			
Plant Parameter		LOFT L3-1 (Note 1)	SRL Gas Pressurizer (Note 2)	KMR-2	
P (MPa)	0.2–5.0	1.5–4.5			
q (W/cm^2)					
G (kg/m^2-s)	0–16100				
Comments		Ref. 1	Ref. 3	Ref. 5	

Nomenclature
P, pressure
q, heat flux
G, mass flux

References
1. P. D. Bayless, J. B. Marlow, and R. H. Averill, “Experimental Data Report for LOFT Nuclear Small-Break Experiment L3-1,” EG&G Idaho, Inc. document NUREG/CR-1145, also EGG-2007 (January 1980).
2. K. E. Carlson, R. A. Riemke, S. Z. Rouhani, R. W. Shumway, and W. L. Weaver, “RELAP5/MOD3 Code Manual, Volume III: Developmental Assessment Problems,” EGG&G Idaho, Inc. Draft document NUREG/CR-5535, also EGG-2596, Volume III (June 1990).
3. W. L. Howarth and R. A. Dimenna, “SRS Supplemental Safety System Injection (Gas Pressurizer) Test,” Westinghouse Savannah River Company report WSRC-MS-92-519 (May 3, 1993).
4. W. L. Howarth and R. A. Dimenna, “RELAP5 MOD3 Analysis of SRS Supplemental Safety System Injection (Gas Pressurizer) Test,” Westinghouse Savannah River Company report WSRC-MS-92-519X (December 29, 1992).
5. A. S. Devkin and B. F. Balunov, “RELAP5/MOD3 Assessment for the Depressurization Processes at the Test Facility KMR-2 with Gas-Steam Pressurizer,” Proceedings of the International Conference on New Trends in Nuclear System Thermohydraulics, Pisa, Italy, Volume 1, pp. 429-33 (May 30 - June 2, 1994).

Notes

- This test was used to validate the accumulator model in RELAP5/MOD3 as described in Ref. 2, Section 2.2.7.
- This test was used to validate the accumulator model in RELAP5/MOD3 as described in Ref. 4.

TABLE F-10
CANDIDATE COMMON EXPERIMENTAL FACILITIES: HEAT CONDUCTANCE–FUEL-CLAD GAP

Plant	Westinghouse AP600		
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)		
Transient Phase	Blowdown		
PIRT Parameter	Gap conductance		
	Plant Range	Test Facility	
Plant Parameter		Modified Pulse Design (low pressure)	Modified Pulse Design (high pressure)
Gas pressure (MPa)	2.5	0.1	
Temperature (K)	294	293 - 873	
Gas composition (Note 1)	Helium (94.7%) Air (4.4%) Argon (0.5%) Xenon (0.34%) Krypton (0.06%)	helium (100), argon (100), xenon, (100) helium / argon (51.8 / 48.2), and helium / xenon (89 / 11)	
Interfacial surface morphology or ISM (μm)		Depleted UO_2: ISM-I =14.4±2.8; ISM-II = 1.6±0.7; and ISM-III = x±0.05 Zircaloy-4: ISM-I =4.5±0.4; ISM-II = 0.4±0.2; and ISM-III = x±0.05	
Gap width (μm)	10	2.7 - 33.0	
Comments	Above as-built conditions	Source of data is Ref. 3. Reference 4 reports use of the data to validate a modified model.	Source of data is Ref. 5.

Plant	Westinghouse AP600		
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)		
Transient Phase	Blowdown		
PIRT Parameter	Gap conductance		

	Plant Range	Test Facility	
Plant Parameter		Power Burst Facility	Halden assembly IFA-226
Gas pressure (MPa)	2.5		
Temperature (K)	294		
Gas composition (Note 1)	Helium (94.7%) Air (4.4%) Argon (0.5%) Xenon (0.34%) Krypton (0.06%)		helium, argon, xenon, krypton, nitrogen, hydrogen
Interfacial surface morphology or ISM (μm)			
Gap width (μm)	10		210 - 250
Comments	Above as-built conditions	Source of data is Ref. 6.	Source of data is Ref. 7, as reported in Ref. 8

Nomenclature
See Plant Parameters

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
3. J. E. Garnier and S. Begej, "Ex-Reactor Determination of Thermal Gap and Contact Conductance Between Uranium Dioxide: Zircaloy-4 Interfaces - Stage I - Low Gas Pressure," Pacific Northwest Laboratories document PNL-2697, NUREG/CR-0330 (January 1979).
4. V. K. Chandola and S. K. Loyalka, "Gap Conductance and Temperature Transients in Modified Pulse Design Experiments," Nuclear Technology, Vol. 56, pp. 434-446 (March 1982).

5. J. E. Garnier and S. Begej, "Ex-Reactor Determination of Thermal Gap and Contact Conductance Between Uranium dioxide: Zircaloy-4 Interfaces - Stage II: High Gas Pressure," Pacific Northwest Laboratories document PNL-2232, NUREG/CR-0330, Vol. 2 (July 1980).
6. G. A. Berna, et al., "Gap Conductance Test Series-2 test Results Report for Tests GC 2-1, GC 2-2, and GC 2-3," NUREG/CR-0300, TREE-1268 (November 1978).
7. E. T. Laats, P. E. MacDonald, and W. J. Quapp, "USNRC-OECD Halden Project Fuel Behavior Test Program - Experiment Data Report for Test Assemblies IFA-226 and IFA-239," Idaho Nuclear Engineering Laboratory (December 1975).
8. P. E. MacDonald and J. Weisman, "Effect of Pellet Cracking on Light Water Reactor Fuel Temperatures," Nuclear Technology, Vol. 31, pp. 357-366 (December 1976).

Notes

1. Gas composition used in B. E. Boyack, et al., "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Methodology to a Large-Break Loss-of-Coolant Accident," EG&G Idaho, Inc. document NUREG/CR-5249, also EGG-2552 (October 1989).
2. See Ref. 1 for a brief description of the current TRAC model, section 3.4.5, pg. 3-85 to 3-86.
3. Gap conductance is not identified as an experimental parameter in Ref. 2.
4. Experimental results show that fuel pellets crack, relocate, and are eccentrically positioned within the sheath. As a result, the heat transfer across the fuel-sheath gap is significantly greater than that which is calculated with fuel pellet modeling as solid concentric cylinder (See Ref. 8).

TABLE F-11
CANDIDATE COMMON EXPERIMENTAL FACILITIES: HEAT TRANSFER–FORCED CONVECTION TO VAPOR

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Forced Convection to Vapor (Note 1)				
	Plant Range	Test Facility			
Plant Parameter		Babus'Haq	Davies &Al-Arabi		
P (MPa)	0.1				
q (W/cm^2)	1				
v (m/s)	0-4				
G (kg/m^2-s)	10-20				
Re (core)	$1.4\text{-}2.8 \times 10^4$	$1.2\text{–}5.5 \times 10^4$			
Comments		Ref. 1: Tests performed with air rather than steam	Ref. 2: Tests performed with water		

Nomenclature
P, pressure
q, heat flux
v, velocity
G, mass flux
Re, Reynolds Number

References

1. R. F. Babus'Haq, "Forced-Convective Heat Transfer from a Pipe to Air Flowing Turbulently Inside It," Experimental Heat Transfer, Vol. 5, pp. 161-173 (1992).
2. V. C. Davies and M. Al-Arabi, "Heat Transfer Between Tubes and a Fluid Flowing Through Them with Varying Degrees of Turbulence Due to Entrance Conditions," Proc. Inst. Mech. Eng, Vol. 169, pp. 993-1006 (1955).

TABLE F-12
CANDIDATE COMMON EXPERIMENTAL FACILITIES: HEAT TRANSFER–STORED ENERGY RELEASE

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Blowdown
PIRT Parameter	Stored Energy Release

	Plant Range	Test Facility	
Plant Parameter		Power Burst Facility Test PCM-2 (Ref. 1-2)	Power Burst Facility Test LOC-11C (Ref. 3-4)
P (MPa)	5.1–15.4	13.53	
q (W/cm^2)	7–46	136	
G (kg/m^2-s)	0–2455	750–1361	
Comments	Above as-built conditions	Unirradiated fuel used.	

	Plant Range	Test Facility	
Plant Parameter		PHEBUS LBLOCA Test 212 (Ref. 5)	LOFT L6-8B-1 and L6-8B-2 (Ref. 6-7)
P (MPa)	5.1–15.4		14.6 rising to 15.7 decreasing to 14.2
q (W/cm^2)	7–46		
G (kg/m^2-s)	0–2455		
Comments	Above as-built conditions	Nuclear fuel rods used.	Fuel centerline temperature available during slow transient with controlled core conditions.

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. Z. R. Martinson, “Power-Cooling-Mismatch test serest test PCM-2 Test Results Report,” Idaho National Engineering Laboratory document NUREG/CR-1038 (1977).
2. R. O. Montgomery, Y. R. Rashid, J. A. George, K. L. Peddicord, and C. L. Lin, “Validation of FREY for the Safety Analysis of LWR Fuel Using Transient Fuel Rod Experiments,” Nuclear Engineering and Design, Vol. 121, pp. 395-408 (1990).
3. J. R. Larson, et al., “PBF-LOCA Test Series, Test LOC-11 Test Results Report,” NUREG/CR-0618, TREE-1329 (March 1979).
4. P. E. MacDonald, J. M. Broughton, and J. W. Spore, “An Evaluation of the Thermal-Hydraulic and Fuel Rod Thermal and Mechnical Behavior During the First Power Burst Facility Nuclear Tests,” Nuclear Technology, Vol. 44, pp. 401-410 (August 1979).
5. M. Reocreus and E. F. Scott de Martionville, “A Study of Fuel Behavior in PWR Design Basis Accident: An Analysis of Results from the PHEBUS and EDGAR Tests,” Nuclear Engineering and Design, Vol. 124, pp. 363-378 (1990).
6. D. B. Jarrell, J. M. Divine, and K. J. McKenna, “Experimental Data Report for LOFT Anticipated Transient Slow and Fast Rod Withdrawal Experiment L6-8,” NUREG/CR0-2930 (July 1982).
7. C. L. Nalezny, “Summary of Nuclear Regulatory Commission’s LOFT Program Experiments,” NUREG/CR-3214 (July 1983).

TABLE F-13
CANDIDATE COMMON EXPERIMENTAL FACILITIES: INTERFACIAL DRAG (CORE AND DOWNCOMER)

Plant	Westinghouse AP600					
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)					
Transient Phase	Reflood					
PIRT Parameter	Core Interfacial Drag					
	Plant Range	Test Facility (Ref. 2)				
Plant Parameter		Dadine	Pericles Rectangular	Pericles Cylindrical	Erset Rod Bundle	
P (MPa)	0.333-0.396	0.1-0.6	0.2-0.4	0.2-0.4	0.1-0.6	
q (W/cm^2)		1-3	2.27-4.36	1.5-4.2	1-7	
Wall Temp. (K)	860-1197	300-600	385-700	355-600	300-900	
Flooding Rate (cm/s)	0-14		0-5	1-19	1-12	
G (kg/m^2-s)	45.5-98.4	20-150	25-50	2-190	10-120	
Subcooling (°C)		20-50	30-90	60	20-80	
Void fraction	0-1.0					
Comments		Heated tube	Rect. 357-rod core	Cylind. 368-rod core	36-rod bundle	

	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		Rebeca	TPTF Jaeri	SCTF Jaeri	CCTF Jaeri
P (MPa)	0.333-0.396	0.2-0.8	3.1-12	≤0.6	≤0.6
q (W/cm^2)					
Wall Temp. (K)	860-1197		≤920K		
Flooding Rate (cm/s)	0-14		≤120		
G (kg/m^2-s)	45.5-98.4		17-94		
Subcooling (°C)			≤20		
Void fraction	0-1.0				
Comments		Critical flow	Horizontal two-phase flow and core heat transfer facility (25-, 24-, and 39-rod core geometries)	2D 8 fuel-rod bundle core	3D 32 fuel-rod bundle core

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Reflood				
PIRT Parameter	Core Interfacial Drag				
	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		Frigg/Froja	Neptun-1 & Neptun-2 Reflood	Achilles Reflood Loop	Thetis Bundle
P (MPa)	0.333-0.396	3-8.7	0.1-0.41		0.13-0.4
q (W/cm^2)		21-89			
Wall Temp. (K)	860-1197		757-867		
Flooding Rate (cm/s)	0-14		1.5-15	4-30	1-6
G (kg/m^2-s)	45.5-98.4	470-2160			
Subcooling (°C)		2-30	11-78		
Void fraction	0-1.0				
Comments		6-rod (Froga) and 36-rod (Frigg) test sections	33 rod test section	68 rod test section ballooned and unballooned tests	7x7 rod test section

	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		Flecht-Seaset/W	THTF/ORNL	G2/W	BCL
P (MPa)	0.333-0.396	0.14-0.41			
q (W/cm^2)					
Wall Temp. (K)	860-1197				
Flooding Rate (cm/s)	0-14	1.5-15			
G (kg/m^2-s)	45.5-98.4				
Subcooling (°C)		3-78			
Void fraction	0-1.0				
Comments		17x17 rod bundle	Ref. 5, Note 4	No info sheet	No info sheet

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
3. C. Unal and R. A. Nelson, "A Phenomenological Model of the Thermal-Hydraulics of Convective Boiling During the Quenching of Hot Rod Bundles Part II: Assessment of the Model with Steady-State and Transient Post-CHF Data," *Nuclear Engineering and Design* **136**, 298-318 (1992).
4. C. Unal, E. Haytcher, and R. A. Nelson, "Thermal-Hydraulics of Convective Boiling During the Quenching of Hot Rod Bundles Part III: Model Assessment Using Winfrith Steady-State Post-CHF Void Fraction and Heat Transfer Measurements and Berkeley Transient Reflood Test Data," *Nuclear Engineering and Design* **140,** 211-227 (1993).
5. D. G. Morris, G. L. Yoder, and C. B. Mullins, "An Experimental Study of Rod Bundle Dispersed-Flow Film Boiling with High-Pressure Water," *Nuclear Technology*, **69**, 82-93 (April 1985).

Notes
1. The CCTF-Run 14 and the Lehigh rod-bundle reflood test 02/24/85-20 were used in Ref. 3 to assess the interfacial drag during reflood.
2. A series of steady-state Winfrith heated tube tests were used in Ref. 4 to assess the axial void-fraction profile.
   - The core interfacial drag has also been indirectly assessed with Flecht-Seaset Tests 31504 and 33436, CCTF Core-II Run 54, and STCF Run 719.
   - Reference 5 is just one of many ORNL documents that must be examined to determine the appropriate tests to be used.

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Downcomer Interfacial Drag				
	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		UPTF	CCTF JAERI	1/15 +1/30 Vessel/Creare	BCL
P (MPa)	0.333-5.06	1-2	≤0.6	0.1-0.45	
Rod Temp. (K)	765-1140				
G (kg/m^2-s)	-357 - 243				
Subcooling (°C)				0-110	
Void fraction	0-1.0				
Comments		1:1 German (KWU) PWR core simultor	3-D 32 fuel-rod bundle core	1/15 and 1/30 vessel downcomer flow tests	No info sheet

Nomenclature

P, pressure

G, mass flux

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).

2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).

TABLE F-14
CANDIDATE COMMON EXPERIMENTAL FACILITIES: LEVEL

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Pipes				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			
Plant Parameter		Vertical Canon	Tapioca	UPTF	Battelle BWR
P (MPa)	0.333-0.396	13	15	1-2	54, 70, 88 bar
q (W/cm^2)		1-3		1.5-4.2	
Wall Temp. (K)	860-1197	300-600		355-600	
Flooding Rate (cm/s)	0-14				
G (kg/m^2-s)	45.5-98.4	20-150		2-190	
Subcooling (°C)		20-50		60	
Void fraction	0-1.0				
Temperature		500-590K	280°C		256-302°C
OECD Facility ID		3.4	3.6	4.1	4.4
Facility Description		Vertical Blowdown, 4.5 m, 0.1 m diam. tube, break at top, 3 to 15 mm diam. Used for TRAC-PF1/MOD1 critical flow assessment	Blowdown facility, 0.324 m ID, 2.6 m length, 0.2144 m^3 volume, break locations at side, top, bottom, middle; break size 2, 5, 10, 20, 35 mm ID	1:1 German (KWU) PWR core simulator	1:80 volume scale of BWR Vessel, 0.6 m ID, to evaluate steam line and feedwater LOCAs, electrical heater, 600kW, 42 heater tube bundle. Discharge nozzle at 6.4, 10.0, 11.2 m height, break diam.: 33, 45, 64, and 76 mm

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Pipes				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			Ref. 4
Plant Parameter		Marviken	Lotus	Single Tube Level Swell	Shoukri Subcooled Boiling
P (MPa)	0.333-0.396	1-5.2	1.7-3.77 bar	0.1	0.15 - 0.17
q (W/cm^2)					
Wall Temp. (K)	860-1197				
Flooding Rate (cm/s)	0-14				
G (kg/m^2-s)	45.5-98.4		4-290 air 5-1000 water		
Subcooling (°C)		0 - 50			
Void fraction	0-1.0	0 - 1.0			
OECD Facility ID		8.2	10.13	10.14	
Facility Description		Large scale critical flow facility. Test T-11 is a level swell experiment with the break located at the top of the vessel (See Ref. 3)	Vertical air-water annular flow tube section, 31.8 mm ID, 20 m length, upflow	Vertical electrically heated tube, steady state level swell tests, 3 m length, 12.5 mm ID, stainless steel	Vertical stainless steel tube, 12.7 mm ID and 30.6 cm length. (See Ref. 4)

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Core				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			
Plant Parameter		Pericles Cylindrical	TPTF Jaeri ROSA IV Program	ECN Reflood and Boildown	FRIGG
P (MPa)	0.333-0.396	0.2-0.4 1.0 - 6.0	3-12 MPa 0.5-12 MPa	2-6 bar	5 MPa
q (W/cm^2)		1.5-4.2 1-2	3-18	1.7-5	
Wall Temp. (K)	860-1197	355-600°C 600°C	≤ 920 K		
Flooding Rate (cm/s)	0-14		≤ 1.2 m/s	1.4-9	
G (kg/m^2-s)	45.5-98.4	1-19 g/cm2 s 1.7 - 3 g/cm2 s	13-98 kg/m2s		
Subcooling (°C)		60°C < 10°C	≤ 20°C	20-80°C	
Void fraction	0-1.0				
OECD Facility ID		3.9	6.1	7.1/7.2	8.3
Facility Description		Cylindrical 368-rod core, 17 x 17 array, for low pressure and high pressure reflooding, also boil-off steady-state and transient tests, 0.95 cm OD, 3.656 m length	Horizontal two-phase flow and core heat transfer facility (25-, 24-, and 39-rod core geometries); Low flow heat transfer tests, boil-off tests, and reflood tests	36 rod test section, 10.7 mm diam, 3 m length, boiloff and reflood tests	6-rod (FROGA) and 36-rod (FRIGG) test sections, Marviken BHWR fuel element design. Extensive number of tests

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Core				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			
Plant Parameter		Neptun-1 Boiloff	Achilles Reflood Loop	Thetis	GE Level Swell
P (MPa)	0.333-0.396	1-5 bar 1-4.1 bar			
q (W/cm^2)		24.6 - 75.1 kW 2.45-4.19 kW/rod			
Wall Temp. (K)	860-1197	757, 867°C			
Flooding Rate (cm/s)	0-14	1.5 - 15			
G (kg/m^2-s)	45.5-98.4				
Subcooling (°C)		0-39°C 11-78°C			
Void fraction	0-1.0				
OECD Facility ID		9.1	10.1	10.2	11.44
Facility Description		33 rod test section, emergency core cooling heat transfer tests in PWR core geometry, boil-off and reflood tests	68 rod test section ballooned and unballooned tests	7x7 test section, PWR core heat transfer during LOCA, reflood tests with clad ballooning blockage, single phase heat transfer tests, level swell tests	Blowdown facility, 14 ft pressure vessel with different size orifice plates to control depressurization

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Liquid Level in Downcomer				
	Plant Range (Note 2)	OECD Test Facility (Ref. 2)			
Plant Parameter		UPTF	CCTF Jaeri	1/15 +1/30 Vessel/Creare	BCL
P (MPa)	0.333-5.06	1-2	≤0.6	0.1-0.45	
q (W/cm^2)					
Wall Temp. (K)	765-1140				
G (kg/m^2-s)	-357 - 243				
Subcooling (°C)				0-110	
Void fraction	0-1.0				
OECD Facility ID		4.1	6.15	11.13	
Facility Description		1:1 German (KWU) PWR core simultor	Full height 3-D 32 fuel-rod bundle core. Each bundle has 57 heater rods, 10.7-mm OD, 3.66-m heated length, 7 nonheated rods 13.8-mm OD, 8x8 square lattice with 14.3-mm pitch, 4 loops with 2 steam generators, 4 pump simulators, ECCS injection in cold legs only	1/15 and 1/30 vessel downcomer flow tests	No info sheet

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
3. M. A. Grolmes, A. Sharon, C. S. Kim, and R. E. Paul, "Level Swell Analysis of the Marviken Test 11," *Nuclear Science and Engineering*, **93** (3), 229-239 (1986).
4. M. Shoukri, B. Donevski, R. L. Judd, and G. R. Dimmick, "Experiments on Subcooled Flow Boiling and Condensation in Annular Channels," in Proceedings of the International Seminar on Phase Interface Phenomena in Multiphase Systems (Hemisphere Publishing, 1991), pp. 413–422.

Notes

1. Plant range shown is for reflood phase of AP600 LBLOCA in core.
2. Plant range shown is for refill phase of AP600 LBLOCA in downcomer.

TABLE F-15
CANDIDATE COMMON EXPERIMENTAL FACILITIES: NONCONDENSIBLE EFFECTS

Plant	Westinghouse 4-Loop PWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Reflood				
PIRT Parameter	Noncondensable Effects				
	Plant Range	Test Facility (Note 1)			
Plant Parameter		MIT Steam Condensation	MIT Single-Tube Experiment	UCB Steam Condensation	
P (MPa)	0.1	1.5–4.5			
Re_i			5000–11400		
F (%)		Air:35–85	Air:10–35 Helium: 2–10		
Comments		Ref. 1	Refs. 2-3	Refs. 4-5	

Nomenclature
P, pressure
Re_I , inlet mixture Reynolds number
F, noncondensable fraction

References

1. Dehbi, M. W. Golay, and M. S. Kazimi, "The Effects of Non-Condensable Gases on the Steam Condensation under Turbulent Natural Convection Conditions," Massachusetts Institute of Technology document MIT-ANP-TR-004 (June 1990).
2. M. Siddique, "The Effects of Noncondensable Gases on Steam Condensation under Forced Convection Conditions," Ph.D. Thesis, Massachusetts Institute of Technology (January 1992).
3. M. Siddique, M. W. Golay, and M. S. Kazimi, "Local Heat Transfer Coefficients for Forced-Convection Condensation of Steam in a Vertical Tube in the Presence of a Noncondensable Gas," Nuclear Technology, Vol. 102, pp. 386-402 (1993).
4. M. Vierow and V. E. Schrock, "Condensation Heat Transfer in Natural Circulation with Noncondensable Gas," Department of Nuclear Engineering, University of California at Berkeley document UCB-NE 4170 (May 1990).
5. S. Z. Kuhn, V. E.Schrock, and P. F. Peterson, "Final Report on U. C. Berkeley Single Tube Condensation Studies," University of California Berkeley document UCB-NE-4201, Rev. 2 (1994).

Notes

1. The MIT steam condensation, MIT single-tube experiment, andUCB steam condensation experiments were previously used for assessing the noncondensable model in RELAP5/MOD3 (Y. A. Hassin and S. Banerjee, “Implementation of a Non-Condensible Model in RELAP5/MOD3,” Nuclear Engineering and Design, Vol. 162, pp. 281-300 (1996).

TABLE F-16
CANDIDATE COMMON EXPERIMENTAL FACILITIES: ASYMMETRIES

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill				
PIRT Parameter	Asymmetric Flow				
	Plant Range	Test Facility (note 2)			
Plant Parameter		LOFT L2-5			
P (MPa)	0.1 – 15.4	0.1 – 14.95			
q (W/cm^2)	0.1 – 46	0.72 – 36.0 MW			
G (kg/m^2-s)	1 – 2455	192.4 kg/s			
Comments		Refs. 1-2			

<u>Nomenclature</u>
P, pressure
q, heat flux
G, mass flux

<u>References</u>

1. C. L. Nalezny, "Summary of Nuclear Regulatory Commission's LOFT Program Experiments," Idaho National Engineering Laboratory document EGG-2248, also NUREG/CR-3214 (July 1983).
2. P. D. Bayless and J. M. Divine, "Experiment Data Report for LOFT Large Break Loss-of-Coolant Experiment L2-5," Idaho National Engineering Laboratory document EGG-2210 also NUREG/CR-2826 (August 1982).

TABLE F-17
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–COUNTERCURRENT

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Countercurrent Flow–Downcomer				
	Plant Range	Test Facility			
Plant Parameter		Dartmouth	Bankoff	BCL	Creare
P (MPa)	0.1			0.1–0.4	
$T_{ECC\ inj}$ (K)				277-366	288–366
G_v (kg/m^2-s)				8.3 lb/s	0–5.5 lb/s
G_l (kg/m^2-s)				575 gpm	0–1500 gpm
Comments		Ref. 1:	Refs.5-6	Ref. 2: Note 1	Ref. 3: Note 2

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Countercurrent Flow–Downcomer				
	Plant Range	Test Facility			
Plant Parameter		UPTF	UPTF		
P (MPa)	0.1				
$T_{ECC\ inj}$ (K)		50 subcooled			
G_v (kg/m^2-s)		100 kg/s			
G_l (kg/m^2-s)		735–1465 kg/s			
Comments		Ref. 4: Test 6 Downcomer	Ref. 4: Test 10C Upper tie plate		

Nomenclature

P, pressure
G, mass flux

References

1. G. B. Wallis, P. C. DeSicyes, P. J. Roselli and J. Lacombe, “Countercurrent Annular Flow Regimes for Steam and Subcooled Water in a Vertical Tube,” Electric Power Research Institute document NP-1336 (January 1980).
2. R. P. Collier, L. J. Flanigan, and J. A.Dworak, “Data Report on ECC Bypass Tests for TRAC Assessment,” Battelle Columbus Laboratories document (July 1980).
3. C. J. Crowley, P. H. Rothe, and R. G. Sam, “1/5 Scale Countercurrent Flow Data Presenation and Discussion,” Creare, Inc. document NUREG/CR-2106 (November 1981).
4. “Test No. 6 Downcomer Countercurrent Flow Test,” 2D/3D Program Upper Plenum test Facility Experimental Data Report, Siemens/KWU document U9 316/89/14 (1989).
5. S. G. Bankoff, R. S. Tankin, M. C. Yuen, and C.L. Hsieh, “Countercurrent Flow of Air/Water and Steam/Water through a Horizontal Performated Plate,” International Journal of Heat and Mass Transfer, Vol. 24, No. 8, pp. 1381-1395 (1981).
6. I. Dilber and S. G. Bankoff, “Countercurrent Flow Limits for Steam and Cold Water through a Horizontal Perforated Plate with Vertical Jet Injection,” International Journal of Heat and Mass Transfer, Vol. 28, No. 12, pp. 2382-2385 (1985).

Notes

1. BCL operated a $1/15^{th}$-scale model at 60 psi and a $2/15^{th}$-scale facility at low pressures.
2. Creare operated several facilities in scales ranging from 1/30 to 1/5.

TABLE F-18
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–MULTIDIMENSIONAL

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Multidimensional flow (note 1)				
	Plant Range	Test Facility (note 2)			
Plant Parameter		OST (note 3)	Rectangular clarifier (note 4)	Slab Core Test Facility (note 5)	Pericles (note 6)
P (MPa)	5.1 - 15.4	0.1–5.0	0.1	0.2	0.2–0.55
q (W/cm^2)	7 - 46		Isothermal		1.35–5.0
G (kg/m^2-s)	2455 - 0		4 - 11 (estimated)		
Comments		Problem has been calculated as reported in Ref. 3.	Data reported in Ref. 4; analysis using data reported in Ref. 5.		

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Multidimensional flow (note 1)				
	Plant Range	Test Facility (note 2)			
Plant Parameter		Cylindrical Core Test Facility (note 7)			
P (MPa)	5.1 - 15.4	0.2			
q (W/cm^2)	7 - 46				
G (kg/m^2-s)	2455 - 0				
Comments					

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
2. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
3. H. Stadtke, G. Franchello, and B. Worth, "Numerical Simulation of Multi-Dimensional Two-Phase Flow Based upon Flux Vector Splitting," Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics NURETH-7, NUREG/CP-0142, Vol. 1, pp. 648-672 (September 10-15, 1995).
4. Imam, "Numerical Modeling of Rectangular Clarifiers," Ph.D. Thesis, University of Windsor (1981). Will request data after it is determined that this is a valid element of the validation test matrix.
5. H. Gerges and J. McCorquodale, "Modelling of Flow in Rectangular Sedimentation Tanks by an Explicit Third-Order Upwinding Technique," International Jouranl for Numerical Methods in Fluids, Vol. 24, pp. 537-561 (1997).
6. B. E. Boyack, P. R. Shire, and S. C. Harmony, "TRAC-PF1/MOD1 Code Assessment Summary Report For SCTF Core-III," Los Alamos National Laboratory restricted distribution document LA-CP-90-71 (February 8, 1990).
7. H. J. Stumpf, "CCTF Run 76 TRAC-PF1/MOD1 Analysis," Los Alamos National Laboratory document LA-2D/3D-TN-86-6 (April 1986).
8. H. J. Stumpf, "CCTF Run 77 TRAC-PF1/MOD1 Analysis," Los Alamos National Laboratory document LA-2D/3D-TN-86-5 (May 1986).

Notes

1. For the blowdown phase, multidimensional phenomena in the core was highly ranked. This phenomenon appears in the OECD/CSNI test matrix (Ref. 1) as Category 10, Global Multidimensional Fluid Temperature, Void and Flow Distribution with the following plant components identified: upper plenum, core, downcomer, and steam-generator secondary side.
2. We have attempted to list the experimental facilities moving from most fundamental separate effect tests to integral tests.
3. Should be considered as "Other Standard Test" or OST in the "concept category," as described in Ref. 2. Problem models the blowdown of a partially filled pressure vessel through a horizontal discharge line.
4. Parameters do not correspond to AP600 blowdown parameters. Should consider this test as basic proof of principle, i.e., used to evaluate the degree to which basic two-dimensional phenomena are calculated in an isothermal condition.
5. Use SCTF Runs 718, 719, 720 which characterize multidimensional core flows with the multidimensionality induced by the radial core power profile. Run 718 has a uniform radial core power profile; Run 719 has 1.36, 1.20, 1.10, 1.00, 0.91, 0.86, 0.81, and 0.76 peak-to-average power ratios across the 8 test assemblies; Run 720 has 0.81, 0.91, 1.1, 1.36, 1.20, 1.00, 0.86, and 0.76 across the 8 test assemblies. All three tests have previously been used for TRAC assessment (See Ref. 6). These tests most directly apply to the refill and reflood phases. SCTF is OECD/CSNI SET facility 6.14 (Ref. 1).

6. Multidimensionality induced by the radial core power profile with the radial peaking factor between 1 and 1.85. These tests most directly apply to reflooding and boiloff. Pericles is OECD/CSNI SET facility 3.8 (Ref. 1).
7. Use CCTF Runs C2-16/76, the base case for the CCTF upper plenum injection tests or C2-18/78, the UPI best estimate case. Both tests have previously been used for TRAC assessment (See Refs. 7-8).

TABLE F-19
CANDIDATE COMMON EXPERIMENTAL FACILITIES: OSCILLATIONS

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Oscillations				
	Plant Range	Test Facility			
Plant Parameter		U-Tube Manometer (Ref. 1)	Frigg Dynamic tests (Refs. 2-4)	Flecht-Seaset (Refs. 5-7)	Slab Core Test Facility (Refs. 8-9)
Plant Parameter					
P (MPa)	0.2	0.1			
q (W/cm^2)					
G (kg/m^2-s)	0.0 - 4150				
Comments	Check core and downcomer flows during refill and enter in plant parameter section	Single phase liquid – analytical solution exists	Tests 662101, 662105, 662107, 662113, 462053, and 462101, See Note 1.	Test 33437 - See Note 2.	Test S2-08 (Run 613). See Note 3

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. R. G. Steinke, “A Description of the Test Problems in the TRAC-P Standard Test Matrix,” Los Alamos National Laboratory document LA-UR-96-1475 (May 1996).
2. O. Nylund, K. M. Becker, R. Eklund, O. Gelius, I. Haga, A. Jensen, D. Maines, A. Olsen, Z. Rouhani, J. Skaug, and F. Akerhielm, “Hydrodynamic and Heat Transfer Measurements on a Full-Scale Simulated 36-Rod BHWR Fuel Element with Non-Uniform Radial Heat Flux Distribution, FRIGG-3,” AB Atomenergi document R4-494/RL-1254 (1970).
3. O. Nylund, K. M. Becker, R. Eklund, O. Gelius, I. Haga, A. Jensen, D. Maines, A. Olsen, Z. Rouhani, J. Skaug and F. Akerhielm, “Hydrodynamic and Heat Transfer Measurements on a Full-Scale Simulated 36-Rod BMWR Fuel Element with Non-Uniform Axial and Radial Heat Flux Distribution, FRIGG-4,” AB Atomenergi document R4-502/RL-1253 (1970).
4. U. S. Rohatgi, L. Y. Neymotin and W. Wulff, “Assessment of RAMONA-3B Methodology with Oscillatory Flow Tests,” Nuclear Engineering and Design, Vol. 143, pp. 69-82 (1993).
5. M. J. Loftus, L. E. Hochreiter, C. E. Conway, C. E. Dodge, A. Tong, E. R. Rosal, M. M. Valkovic, and S. Wong, “PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report,” US Nuclear Regulatory Commission document NUREG/CR-1532, Electric Power Research Institute document EPRI NP-1459, Westinghouse Electric Corporation document WCAP 9699 (June 1980).
6. N. Lee, S. Wong, H. C. Yeh, and L. E. Hochreiter, “PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Evaluation and Analysis Report,” US Nuclear Regulatory Commission document NUREG/CR-2256, Electric Power Research Institute document EPRI NP-2013, Westinghouse Electric Corporation document WCAP 9891 (September 1981).
7. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore, and T. D. Knight, “Reflood Completion Report, Volume II, Developmental Assessment of a New Reflood Model for the TRAC-M/F77 Code,” Los Alamos National Laboratory document LA-UR-98-3043 (April 1998).
8. Y. Abe, T. Oyama, M. Sobajima, I. Arase, Y. Niitsuma, T. Iwamura, K. Nakajima, T. Chiba, K. Komori, H. Sonobe, T. Kuroyangi, A. Owada, H. Adachi, C. E. Winsel, and Y. Murao, “Data Report of Large Scale Reflood Test – 75 – SCTF Test S2-08 (Run 613),” Japan Atomic Energy Research Institute document JAERI-memo 59-437 (February 1985).
9. J. C. Lin, “TRAC-PF1 Calculation of SCTF Core-II Flecht Seaset Coupling Test S2-08 (Run 613),” Los Alamos National Laboratory document LA-2D/3E-TN-85-2 (February 1985).
10.

Notes

1. Rohatgi, et al., used the FRIGG data of Refs. 2-3 to assess the RAMONA-3B code. The oscillations are externally induced by core power variations. The geometry includes a downcomer and core connected by a horizontal pipe. FRIGG is closer to a SET than IET and it appears that the FRIGG data is a good candidate for assessment of the code’s capability to predict oscillatory phenomena measured in a facility with two-phase flow that is simpler than IET facilities.
2. Current TRAC input deck exists and was used in the assessment reported in Ref. 7.
3. Test has previously been assessed for TRAC-PF1 as reported in Ref. 9.

TABLE F-20
CANDIDATE COMMON EXPERIMENTAL FACILITIES: POWER–DECAY HEAT

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, Reflood, Long-Term Cooling				
PIRT Parameter	Decay Heat				
	Plant Range	Test Standard			
Plant Parameter		ANS-5.1-1994 (Ref. 1)	AESJ (Ref. 2-3)	ISO (Ref. 4)	
T (s)	$0.0 - 10^{10}$	$0.0 - 10^{10}$	$0.0 - 10^{10}$	$0.0 - 10^{10}$	
Comments		American National Standard	Proposed Japanese Standard	Proposed International Standards Organization Standard	

Nomenclature
T, Time

References

12. "American National Standard: For Decay heat Power in Light Water Reactors," American Nuclear Society standard ANSI / ANS-5.1-1979(R1985) (1985).
13. K. Tasaka, T. Katoh, J. Katakura, T. Yosida, S. Iijima, R. Nakasima and S. Nagayama, "Summary Report – Recommendation on Decay Heat Power in Nuclear Reactors," Journal of Nuclear Science and Technology, Vol. 28, No. 12, pp. 1134-1142 (December 1991).
14. K. Tasaka, et al., "Recommended Values of Decay Heat Power and Method to Utilize the Data," Japan Atomic Energy Research Institute document JAERI-M 91-034 (1991).
15. "Nuclear Energy-Light Water Reactors-Calculation of the Decay Heat Power in Nuclear Fuels," International Organization for Standardization standard ISO / DIS 10645 (1990).

TABLE F-21
CANDIDATE COMMON EXPERIMENTAL FACILITIES: PUMP PERFORMANCE, INCLUDING DEGRADATION

Plant	Westinghouse 4-Loop PWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Pump Degradation				
	Plant Range	Test Facility			
		Semiscale	EPRI	KWU	
P (MPa)	5.1–15.4				
Head (m) (Note 1)			~95	92	
Specific Speed		18	82	130	
Comments		Ref. 1: Pump is of the radial-flow type.	Ref. 2: Pump is of the mixed-flow type.	Ref. 3: Pump is of the axial-flow type used in KWU reactors.	

Nomenclature
P, pressure
G, mass flux

References

1. D. J. Olson, “Experiment Data Report for Single and Two-Phase Steady State Tests of the 1-1/2 Loop Mod1-1 Semiscale System Pump,” Westinghouse Canada Ltd. Document ANCR-1150 (May 1974).
2. “Pump Two-Phase Performance Program,” Electric Power Research Institute document EPRI NP-1556, Volumes 1-8 (September 1980).
3. W. Kastner and G. J. Seeberger, “Pump Behavior and Its Impact on a Loss-of-Coolant Accident in a Pressurized Water Reactor,” Nuclear Technology, Vol. 60, pp. 268-277 (February 1983).

Head
1. Steady-state design point single-phase head.

TABLE F-22
CANDIDATE COMMON EXPERIMENTAL FACILITIES: REACTIVITY–VOID

This page intentionally left blank.

No tests identified.

APPENDIX G

EXPANDED LISTING OF TRAC-M INPUT DECKS FOR COMMON AND PWR-SPECIFIC SETS, IETS AND PLANTS

Table G-1 lists the available common and PWR-specific TRAC-M SETs input decks. For each facility input deck, a brief description of the facility, test type, test number, and report reference in addition to the latest code version on which the input deck was exercised are provided. Table G-2 lists the available common and PWR TRAC-M IET input decks in the same format. Table G-3 lists the available PWR TRAC-M plant input decks in the same format.

TABLE G-1
TRAC-M INPUT DECKS FOR SEPARATE EFFECT TESTS

Facility	Type of Test	Test ID	References	Input Deck	Comments
Akimoto	Condensation		G-1	TRAC-M / F77, Version 5.5	Equivalent to MOD2 input deck
Bankoff	CCFL		G-1	TRAC-M / F77, Version 5.5	Air-water and steam-water. Equivalent to MOD2 input deck
BCL	Downcomer counter-current flow	26204, 26502, 29111	G-2	PD2	Deck stored in LANL TRAC Input Deck Archive (TIDA)
Bennett	Heated-tube CHF	5336, 5431, and 5442	G-1	TRAC-M / F77, Version 5.5	Equivalent to MOD2 input deck
Berkeley	Reflood heat transfer		1991 Dev. Assessment	Early MOD2	Deck stored in LANL TIDA
CISE	Critical Flow	4	G-3	PD2	Deck stored in LANL TIDA
CREARE	Downcomer counter-current flow		G-3	Listings in Appendix F	No deck found
Dartmouth	Air-water counter-current flow	2-in pipe and 6-in. pipe	G-3	MOD1	Deck stored in LANL TIDA
Edwards	Critical Flow		G-3	MOD2	Deck stored in LANL TIDA
FLECHT	Reflood heat transfer	4831 17201	G-3	PD2	No deck found
FLECHT-SEASET	Reflood heat transfer	31504 33436	G-1	TRAC-M / F77, Version 5.5	Forced and gravity reflood tests. Equivalent to MOD2 input deck

TABLE G-1 (cont)
TRAC-M INPUT DECKS FOR SEPARATE EFFECT TESTS

Facility	Type of Test	Test ID	References	Input Deck	Comments
Lehigh	Reflood heat transfer		G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
Marviken	Critical Flow	4, 13, 20, 22, and 24	G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
THETIS	Boildown level-swell test		1991 Dev. Assessment	Current MOD2	Deck stored in LANL TIDA
THTF	Rod-bundle blowdown heat transfer	177	G-3	PD2	Deck stored in LANL TIDA
Winfrith	Heated tube CHF		1991 Dev. Assessment	MOD2	Deck stored in LANL TIDA
Moby-Dick	Critical flow	403, 408, 455, 79, 172	G-4	MOD1	Typical input data deck in report
Super-Moby-Dick	Critical flow	1-15	G-4	MOD1	Typical input deck in report
Cannon	Blowdown	D, L, I	G-4	MOD1	Typical input deck in report
Super-Canon	Blowdown	P, X, Q	G-4	MOD1	Typical input deck in report
Vertical-Canon	Blowdown	9, 22, 24	G-4	MOD1	Typical input deck in report
Omega-Tube	Blowdown	3, 6, 8, 9, 29, 30	G-4	MOD1	Typical input deck in report
Omega-Bundle	Blowdown	2, 3, 9, 11, 13, 18, 19	G-4	MOD1	Typical input deck in report
Strathclyde	Refill phase LB LOCA	B/B2; C/C2; D/D2	G-5	MOD1	1/10th scale model of a PWR downcomer
Achilles	Forced/gravity reflood	23, 28	G-6	MOD1	Typical input deck in report
UPTF	LOCA loop flow pattern LOCA downcomer flow pattern	8b 6	G-1	TRAC-M/F77, Version 5.5	Cold-leg flow and downcomer tests. Equivalent to MOD2 input deck Deck stored in LANL TIDA
CCTF	LOCA refill and reflood	14	G-1	TRAC-M/F77, Version 5.5	Direct ECC water injection into lower plenum. Equivalent to MOD2 input deck

TABLE G-2
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
PKL	Natural circulation	ID1-4	G-7	MOD1	Deck stored in LANL TIDA
	Natural circulation	ID1-9	G-7	MOD1	Deck stored in LANL TIDA
	Reflux cooling	ID1-14	G-7	MOD1	Deck stored in LANL TIDA
	Gravity reflood	K9	G-2	MOD1	Deck stored in LANL TIDA
	Gravity reflood	K5.4A	G-2	MOD1	Deck stored in LANL TIDA
Semiscale Mod-1	200% cold-leg break without ECCS	S-02-8	G-3	MOD1	Deck stored in LANL TIDA
	200% cold-leg break with ECCS	S-06-3	G-3	MOD2	Deck stored in LANL TIDA
Semiscale Mod-3	2.5% cold-leg break, early pump trip	S-SB-P1	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break, delayed pump trip	S-SB-P2	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break, late pump trip	S-SB-P7	G-7	MOD1	Deck stored in LANL TIDA
	10% cold-leg break with delayed ECCS and secondary blowdown	S-07-10D	G-7	MOD1	Deck stored in LANL TIDA
	2.5% hot-leg break, pumps off	S-SB-P3	G-2	MOD1	Deck stored in LANL TIDA
	2.5% hot-leg break, pumps on	S-SB-P4	G-2	MOD1	Deck stored in LANL TIDA
	Natural circulation	S-NC-2B	G-8	MOD1	Deck stored in LANL TIDA
	Natural circulation	S-NC-5	G-8	MOD1	Deck stored in LANL TIDA
	Natural circulation	S-NC-6	G-8, G-9	MOD1	Input listing in Reference G-10 Deck stored in LANL TIDA
	Natural circulation	S-NC-7C		MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
Semiscale Mod-2a	10% cold-leg break with upper-head injection (UHI)	S-UT-2	G-8	MOD1	Deck stored in LANL TIDA
	5% cold-leg break without UHI	S-UT-6	G-8	MOD1	Input listing in Reference G-10 Deck stored in LANL TIDA
	5% cold-leg break with UHI	S-UT-7	G-8	MOD1	Input listing in Reference G-10 Deck stored in LANL TIDA
LOFT	Isothermal DEGB blowdown	L1-4	G-3	MOD1	Deck stored in LANL TIDA
	50% power, DEGB cold-leg break	L2-2	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break in broken cold leg	L3-1	G-2	MOD1	Deck stored in LANL TIDA
	15% cold-leg break in broken cold leg	L3-7	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break in intact cold leg, early pump trip	L3-5	G-2	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break in intact cold leg, late pump trip	L3-6	G-2	MOD1	Deck stored in LANL TIDA
	20.7% cold-leg break in broken cold leg	L5-1	G-2	MOD1	Deck stored in LANL TIDA
	20.7% cold-leg break in broken cold leg with delayed ECCS	L8-2	G-2	MOD1	Deck stored in LANL TIDA
	200% cold-leg break, pumps on	L2-3	G-2	MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
	200% cold-leg break, early pump trip	L2-5	G-2	MOD1	Deck stored in LANL TIDA
	200% cold-leg break, higher power	L2-6 (LP-02-6)	G-1	TRAC-M / F77, Version 5.5	Equivalent to MOD2 input deck
	Loss of feedwater transient	L9-1 / L3-3	G-8	MOD1	Deck stored in LANL TIDA
	Cooldown transient	L6-7 / L9-2	G-8	MOD1	Deck stored in LANL TIDA
	Loss of steam load	L6-1	G-1	TRAC-M / F77, Version 5.5	Equivalent to MOD2 input deck
	Pump trip	L6-2	G-9	MOD1	Input listing in Reference
	Excessive-load increase	L6-3	G-9	MOD1	Input listing in Reference
Crystal River Transient	Anticipated transients—non-nuclear instrumentation failure		G-8	MOD1	Deck stored in LANL TIDA
CCTF	Core-I reflood base case	14		MOD1	Deck stored in LANL TIDA
	Core-II reflood low power	54	G-1	TRAC-M / F77, Version 5.5	Equivalent to MOD2 input deck
	Core-II upper plenum injection	57	G-10	MOD1	Deck stored in LANL TIDA
	Core-II upper plenum injection	59	G-10	MOD1	Deck stored in LANL TIDA
	Core-II upper plenum injection	72	G-10	MOD1	Deck stored in LANL TIDA
	Refill / reflood with asymmetric injection	76	G-10	MOD1	Deck stored in LANL TIDA
	Refill / reflood with UPI	78	G-10	MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
	Downcomer injection / vent valves closed	58	G-11	MOD1	Deck stored in LANL TIDA
	Cold- and hot-leg injection	79	G-12	MOD1	Deck stored in LANL TIDA
	Best-estimate	71	G-13	MOD1	Deck stored in LANL TIDA
MIST	Delayed HPI / PORV feed-and-bleed cooling	330302	G-14	MOD1	Deck stored in LANL TIDA
	50-cm^2 SBLOCA	320201	G-15	MOD1	Deck stored in LANL TIDA
	10-cm^2 SBLOCA	3109AA	G-16	MOD1	Deck stored in LANL TIDA
	STGR	3404AA	G-17	MOD1	Deck stored in LANL TIDA
ROSA-IV LSTF	Single- and two-phase natural circulation	ST-NC-02	G-18	MOD1	Deck stored in LANL TIDA
SCTF	refill / reflood	S2-SH2 (Run 605)	G-19	MOD1	Deck stored in LANL TIDA
		OS1	G-20	MOD1	Deck stored in LANL TIDA
		S3-9 (Run 713)	G-21	MOD1	Deck stored in LANL TIDA
		(Run 704)	G-22	MOD1	Deck stored in LANL TIDA
		(Run 714)	G-23	MOD1	Deck stored in LANL TIDA
		S2-03 (Run 608)	G-24	MOD1	Deck stored in LANL TIDA
		S2-08 (Run 613)	G-25	MOD1	Deck stored in LANL TIDA
		S2-09 (Run 614)	G-26	MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
		S2-SH1 (Run 604)	G-27	MOD1	Deck stored in LANL TIDA
		S2-12 (Run 617)	G-28	MOD1	Deck stored in LANL TIDA
		Run 605	G-29	MOD1	Deck stored in LANL TIDA
		S2-06 (Run 611)	G-30	MOD1	Deck stored in LANL TIDA
		S3-15 (Run 719)	G-1	TRAC-M / F77, Version 5.5	Equivalent to MOD2 input deck
UPTF		17	G-31	MOD2	Deck stored in LANL TIDA
		21	G-32	MOD2	Deck stored in LANL TIDA
		27	G-33	MOD2	Deck stored in LANL TIDA
LOBI	1% cold-leg SB LOCA	A2-81	G-34	MOD1	ISP-18 exercise Deck stored in LANL TIDA
	3% cold-leg SB LOCA	BL-02	G-35	MOD1	

TABLE G-3
TRAC-M INPUT DECKS FOR NUCLEAR POWER PLANTS

Vendor Plant	Transient or Accident	References	Decks	Comments
Westinghouse				
AP600	LB LOCA	G-36	TRAC-M/F77	Deck stored in LANL TIDA
CSAU Plant	LB LOCA for code scaling, applicability, uncertainty (CSAU)	G-37	MOD1	Deck stored in LANL TIDA
R. E. Ginna	Steam generator tube rupture	G-38	MOD2	Deck stored in LANL TIDA
H. B. Robinson	SB LOCA, steam generator tube rupture	G-39	MOD2	Deck stored in LANL TIDA
South Texas Project	SB LOCA	G-40	MOD2	Deck stored in LANL TIDA
USPWR 15x15 fuel	LB LOCA	G-41	MOD1	Deck stored in LANL TIDA
USPWR 17x17 fuel	LB LOCA	G-42	MOD1	Deck stored in LANL TIDA
Zion-1	Main feed-line break/loss of feedwater	G-43	MOD2	Deck stored in LANL TIDA
CE				
Arkansas Nuclear One-2	Turbine trip transient	None	MOD1	Deck stored in LANL TIDA. Converted from a RETRAN deck.
Calvert Cliffs-1	Loss of offsite power	G-44	MOD2	Deck stored in LANL TIDA
B&W				
Bellefonte	Steady state only	G-45	MOD2	Deck stored in LANL TIDA
Crystal River	Plant transient of February 26, 1980	G-8	MOD1	Deck stored in LANL TIDA
Davis-Besse	Loss of feedwater	G-46	MOD2	Deck stored in LANL TIDA
Oconee-1	SB LOCA	G-47	MOD2	Deck stored in LANL TIDA
Three Mile Island-2	SB LOCA (TMI-1 accident)	G-48	MOD2	Deck stored in LANL TIDA

References

G-1. B. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore, and J. L. Steiner, "TRAC-M/F77, Version 5.5 Developmental Assessment Manual, Volume I: Assessment Sections not including 2D/3D Tests," Los Alamos National Laboratory document LA-UR-99-6480 (December 1999).

G-2. T. D. Knight (editor), "TRAC-PD2 Independent Assessment," Los Alamos National Laboratory document NUREG/CR-3866 (December 1984).

G-3. T. D. Knight (editor), "TRAC-PD2 Developmental Assessment," Los Alamos National Laboratory document NUREG/CR-3208 (January 1985)

G-4. B. Spindler and M. Pellissier, "Assessment of TRAC-PF1/MOD1 Version 14.3 Using Separate Effects Critical Flow and Blowdown Experiments," Commission of Atomic Energy CENG/SETH document NUREG/IA-0023, Vol. 1 (January 1990).

G-5. W. M. Dempster, A. M. Bradford, T. M. S. Callander, and H. C. Simpson, "An Assessment of TRAC-PF1/MOD1 Using Strathclyde 1/10 Scale Model Refill Tests 2nd Report," University of Strathclyde/Central Electricity Research Laboratories International Agreement Report, NUREG/IA-0053 (March 1992).

G-6. M. Blackmore, "TRAC-PF1/MOD1 Modelling of the Achilles Rig," Winfrith United Kingdom Atomic Energy Authority document AEEW-M 2418 (Not for Publication) (April 1987).

G-7. B. E. Boyack (editor), "TRAC-PF1 Developmental Assessment, Los Alamos National Laboratory document NUREG/CR-3280 (July 1983).

G-8. T. D. Knight and V. B. Metzger (editors), "TRAC-PF1 Independent Assessment," Los Alamos National Laboratory document NUREG/CR-3646 (October 1985).

G-9. M. S. Sahota and F. L. Addessio, "TRAC-PF1/MOD1 Developmental Assessment," Los Alamos National Laboratory document NUREG/CR-4278 (August 1985).

G-10. M. W. Cappiello, H.J. Stumpf, and B. E. Boyack, "CCTF Core-II Upper Plenum Injection Summary," Los Alamos National Laboratory document LA-2D/3D-TN-86-16 (1986).

G-11. D. A. Siebe and B. D. Boyer, " The Analysis of CCTF Run 58 with TRAC-PF1/MOD1," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-86-19 (April 1988).

G-12. M. W. Cappiello, " TRAC-PF1/MOD1 Analysis of CCTF Combined Injection Test Run 79," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-86-20 (December 1986).

G-13. H. Stumpf and G. J. Wilcutt, " CCTF Run 71 TRAC-PF1/MOD1 Analysis," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-86-8 (May 1986).

G-14. B. E. Boyack, "Posttest Analysis of MIST Test 330302 Using TRAC-PF1/MOD2," Los Alamos National Laboratory document NUREG/CR-5889 (September 1992).

G-15. D. A. Siebe, J. L. Steiner, and B. E. Boyack, “Posttest Analysis of MIST Test 320201 Using TRAC-PF1/MOD1,” Los Alamos National Laboratory document NUREG/CR-5888 (September 1992).

G-16. J. L. Steiner, D. A. Siebe, and B. E. Boyack, “Posttest Analysis of MIST Test 3109AA Using TRAC-PF1/MOD1,” Los Alamos National Laboratory document NUREG/CR-5887 (September 1992).

G-17. D. A. Siebe, J. L. Steiner, and B. E. Boyack, “Posttest Analysis of MIST Test 3404AA Using TRAC-PF1/MOD1,” Los Alamos National Laboratory document LA-UR-92-2699 (August 1992).

G-18. H. J. Stumpf and F. E. Motley, " Results of TRAC Analysis of Run ST-NC-02 From the Large-Scale Test Facility," Los Alamos National Laboratory document LA-CP-87-131 (1987).

G-19. P. R. Shire, " TRAC-PF1/MOD1 Re-Analysis of SCTF Core-II Test S2-SH2 (Run 605)", Los Alamos National Laboratory, LA-CP-87-103 (December 1986).

G-20. J. S. Gilbert, “TRAC-PF1/MOD1 Calculation of Operational Study No. 1,” Los Alamos National Laboratory document LA-CP-87-112 (1987).

G-21. P. R. Shire, " TRAC-PF1/MOD1 Analysis of SCTF Core-III Test S3-02 (Run 713)," Los Alamos National Laboratory document LA-CP-88-11 (January 1988).

G-22. B. E. Boyack and P. L. Mascheroni, " A Posttest Analysis of SCTF Run 704 Using TRAC-PF1/MOD1," Los Alamos National Laboratory document LA-CP-88-131 (June 1988).

G-23. S. C. Harmony and B. E. Boyack, " A Posttest Analysis of SCTF Run 714 Using TRAC-PF1/MOD1," Los Alamos National Laboratory document LA-CP-88-234 (September 1988).

G-24. J. C. Lin, " TRAC-PF1/MOD1 Calculation of SCTF Core-II Supply Test S2-03 (Run 608)," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-85-13 (July 1985).

G-25. J. C. Lin, "TRAC-PF1/MOD1 Calculation of SCTF Core-II FLECHT-SET Coupling Test S2-08 (Run 613)," Los Alamos National Laboratory document LA-2D/3D-TN-85-2 (February 1985).

G-26. J. Gilbert, " TRAC-PF1/MOD1 Calculation of SCTF Core II Test S2-09 (Run 614)," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-85-4 (March 1985).

G-27. J. Gilbert, " TRAC-PF1/MOD1 Calculation of SCTF Core II Test S2-SH1 (Run 604)," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-85-6 (March 1985).

G-28. J. Gilbert, " TRAC-PF1/MOD1 Calculation of SCTF-II Test S2-12 (Run 617)," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-86-13 (March 1987).

G-29. P. R. Shire and B. E. Boyack, " Upper Plenum Studies of SCTF Run 605," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-86-15 (August 1986).

G-30. J. Gilbert, " TRAC-PF1/MOD1 Calculation of SCTF Core II Test S2-06 (Run 611)," Los Alamos National Laboratory Group N-9 document LA-2D/3D-TN-86-9 (March 1986).

G-31. D. A. Siebe and H. J. Stumpf, “Assessment of the Ability of TRAC-PF1/MOD1 and TRAC-PF1/MOD2 to Predict the Phenomena Related to Steam-Binding,” Los Alamos National Laboratory document LA-CP-92-79 (March 1992).

G-32. E. Mullen, H. J. Stumpf, D. A. Siebe, “Summary of Downcomer Injection Phenomena for UPTF and TRAC Posttest Analyses,” Los Alamos National Laboratory document LA-CP-92-188 (March 1992).

G-33. D. A. Siebe and H. J. Stumpf, “An Assessment of TRAC-PF1/MOD2 Against Data from the Upper Plenum Test Facility for Accumulator Nitrogen Surge,” Los Alamos National Laboratory document LA-CP-92-408 (November 1992).

G-34. C. G. Richards, “Blind and Post-Test Calculations of ISP18 with TRAC-PF1/MOD1,” United Kingdom Atomic Energy Authority Winfrith [Not for Publication] document AEEW-R 2013 (March 1986).

G-35. C. G. Richards, “Pre-Test Calculation of LOBI Test BL-02 Using TRAC-PF1/MOD1,” United Kingdom Atomic Energy Authority Winfrith [Not for Publication] document AEEW-M 2416 (February 1987).

G-36. B. E. Boyack, J. F. Lime, D. A. Pimentel, J. W. Spore, and T. D. Knight, “Reflood Completion Report, Volume II, Developmental Assessment of a New Reflood Model for the TRAC-M/F77 Code,” Los Alamos National Laboratory document LA-UR-98-3043 (April 1998).

G-37. Technical Program Group, “Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break Loss-of-Coolant Accident,” EG&G Idaho Inc. document NUREG/CR-5249 (October 1989).

G-38. D. A. Siebe, “Conversion and Validation of a R. E. Ginna Input Deck to TRAC-PF1/MOD2 Version 5.4.05,” Los Alamos National Laboratory document LA-CP-95-185 (1995).

G-39. J. F. Lime, “Deck Conversion of the H. B. Robinson-2 Plant Model from TRAC-PF1/MOD1 to TRAC-PF1/MOD2,” Los Alamos National Laboratory document LA-CP-96-0109 (May 1996).

G-40. J. F. Lime, “Deck Conversion of the South Texas Project Plant Model from TRAC-PF1/MOD1 to TRAC-PF1/MOD2,” Los Alamos National Laboratory document LA-CP-96-0026 (January 1996).

G-41. P. R. Shire and J. W. Spore, “TRAC-PF1/MOD1 Analysis of a Minimum-Safeguards Large-Break LOCA in a US/Japanese PWR with Four Loops and 15x15 Fuel,” Los Alamos National Laboratory document LA-CP-87-61 (1987).

G-42. P. Coddington and F. Motley, “TRAC-PF1/MOD1 Analysis of a Minimum-Safeguards Large-Break Loss-of-Coolant Accident in a 4-Loop PWR with 17x17 Fuel,” Los Alamos National Laboratory document LA-2D/3D-TN-86-3 (August 1986).

G-43. S. B. Rodriguez, "Conversion and Validation of a Zion-1 Input Deck to TRAC-PF1/MOD2, Version 5.4.05," Los Alamos National Laboratory document LA-CP-95-214 (1995).

G-44. J. F. Lime, "Deck Conversion of the Calvert Cliffs-1 Plant Model from TRAC-PF1/MOD1 to TRAC-PF1/MOD2," Los Alamos National Laboratory document LA-CP-99-8 (December 1998).

G-45. J. F. Lime, "Deck Conversion of the Bellefonte Plant Model from TRAC-PF1/MOD1 to TRAC-PF1/MOD2," Los Alamos National Laboratory document LA-CP-99-71 (March 1999).

G-46. J. F. Lime, "TRAC-PF1/MOD2 Deck Conversion of the Davis-Besse Plant Model," Los Alamos National Laboratory document LA-CP-95-212 (1995).

G-47. J. F. Lime, "Deck Conversion of the Oconee-1 Plant Model from TRAC-PF1/MOD1 to TRAC-PF1/MOD2," Los Alamos National Laboratory document LA-CP-99-32 (February 1999).

G-48. S. B. Rodriguez, "Conversion and Validation of a Three Mile Island Input Deck to TRAC-PF1/MOD2, Version 5.4.05," Los Alamos National Laboratory document LA-CP-95-183 (1995).

APPENDIX H
RECOMMENDED TESTS FOR THE BWR LBLOCA VALIDATION TEST MATRIX

In this appendix, we present the experimental facilities recommended for the TRAC-M BWR LBLOCA validation test matrix. For each PIRT local-level (LL), component-level (CL), and system-level (SL) LBLOCA phenomenon identified in Section 4 (Table 4-5), but not addressed in the recommended tests for the TRAC-M common LBLOCA validation test matrix (Appendix F), we provide a table identifying recommended tests.

Additional tables are provided for several phenomena covered in Appendix F for which additional BWR-specific tests are recommended.

Each table lists the experimental facilities that have produced data that are recommended for inclusion in the validation test matrix.

Local-level PIRT phenomena are covered in Tables H-1 through H-12. Component- and system-level PIRT phenomena are covered in Tables H-13 through H-26.

TABLE H-1
PROPOSED BWR EXPERIMENTAL FACILITIES: BOILING–FILM

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Boiling–Film**				
	Plant Range	Test Facility (See Also Table F-1 for additional tests)			
Plant Parameter		THTF Film Boiling Tests 3.03.6AR 3.06.6B & 3.08.6C			
P (MPa)	0.3 – 5.0	5.17 - 12.4			
Heat Flux (kw/m2)		160 – 1100			
Equil.Quality (%)	0.1-90	0.15 – 100			
Clad Temps (K)	500 – 1400	600 – 1000			
Mass Flux kg/s-m2		129- 1090			
Comments					

References

1. D. G. Morris, et. al., “An Analysis of Transient Film Boiling of High Pressure Water in a Rod Bundle,” NUREG/CR-2469, ORNL, March 1982.

TABLE H-2
PROPOSED BWR EXPERIMENTAL FACILITIES: BOILING–NUCLEATE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Long-term cooling				
PIRT Parameter	**Boiling–Nucleate**				
	Plant Range	Test Facility			
Plant Parameter		ORNL Test 3.07.9N			
P (MPa)	0.1 – 7.0	12.7			
Wall Superheat (K)	0 – 10	14 - 17			
Void Fraction	0 - 0.4	0.17 - 0.89			
Mass Flux (kg/m2-s)	0 – 1500	806			
Heat Flux (MW/m2)	0 - 0.555	0.94			
Subcooling (K)	10 – 60	14.29			
Comments					

Refererences

1. G. L. Yoder et al., Dispersed Flow Film Boiling in Rod Bundle Geometry Steady State Heat Transfer Data and Correlation Comparisons, NUREG/CR-2456, ORNL-5848, March 1982.

TABLE H-3
PROPOSED BWR EXPERIMENTAL FACILITIES: CONDENSATION–INTERFACIAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood				
PIRT Parameter	**Condensation–interfacial: ECC Water**				
	Plant Range	Test Facility (See Table F-3)			
Plant Parameter					
Pressure (MPa)	0.1- 5.0				
Void fraction	0.0 -1.0				
ECC Temp (F)	80 – 180				

TABLE H-4
PROPOSED BWR EXPERIMENTAL FACILITIES: DRYOUT–CHF

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Dryout–critical heat flux (CHF)**				
	Plant Range	Correlations			
Plant Parameter		Biasi	CISE	Zuber	
Pressure (MPa)	0.1 – 5.0	0.1-14.2	7.0	0.1 – 5.0	
Mass Flux(kg/m^2-s)	0 – 6000	100- 6000	300 -1400	< 100	
Quality	0.1 – 1.0	0.2 – 1.0			
Void	0.7 – 1.0				
Comments				Zuber is applied if flow is countercurrent	

<u>References</u>

1. L. Biasi, et al, " Studies on Burnout: Part 3," Energ. Nucl. 14, 1967, pp.530-536.
2. CISE: Heat Transfer Crisis in Steam-Water Mixtures, Energ. Nucl. 12, 1965
3. N. Zuber et al," The Hydrodynamic Crisis in Pool Boiling of Saturated and Subcooled Liquids," Int. Developments in Heat Transfer, 2, 1961, pp. 230-236.

TABLE H-5
PROPOSED BWR EXPERIMENTAL FACILITIES: FLASHING–INTERFACIAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Flashing–interfacial: lower plenum, core, and downcomer**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Tests 901, 902, 924, 926, 905	FIST Test 6DBA1B		
Pressure (MPa)	0.1 – 5.0	0.1 – 7.0	0.1 – 7.0		

References

1. Tasaka, et al., ROSA-III Double-Ended Break Test series for a Loss-of-Coolant Accident in a BWR," Nucl. Tech. Vol. 68, Jan 1985, pp. 77-93.
2. H. Kumamaru, et al., "Similarity Study of ROSA-III and FIST Large Break Counterpart Tests to BWR Large Break LOCA," Nucl. Engr. and Design 103, pp. 223-238, June 1986.

TABLE H-6
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–CRITICAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	**Flow–Critical**				
	Plant Range	Test Facility (See Table F-8)			
Plant Parameter					
Pressure (MPa)	0.7 – 7.0				
L/D	1 – >10				
Subcooling (K)	0 – 20				

TABLE H-7
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT CONDUCTANCE–FUEL-CLAD GAP

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Heat conductance–fuel-clad gap**	Governs temperature distribution and removal of stored heat			
	Plant Range	Test Facility (See Table F-10)			
Key Physical Parameter					
Pellet:					
k (W/m-K)	(7.5-18.5)10E+3				
T (K)	>530				
Gap:					
h (W/sq.m-K)	(3.3-13.1)10E+3				
Burnup (MWD/T)	0-40,000				
Comments					

TABLE H-8
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT TRANSFER–FORCED CONVECTION TO VAPOR

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood				
PIRT Parameter	**Heat transfer–forced convection to vapor**				
	Plant Range	Test Facility (See also Table F-11)			
Plant Parameter		THTF Bundle Uncovery Tests 3.09.10 I,J, K, L, M, & N	G-2 336 Rod bundle Uncovery Tests 718, 722, 727, &731		
Pressure (MPa)	0.1 – 5.0	3.9 – 7.0	0.1 – 5.5		
Void fraction	1.0	1.0	1.0		
Clad Temp (F)	500- 2200	500 – 1500	500 – 1600		
Vapor Temp (F)	500 –1800	500 – 1200	500 -1300		
Vapor Re	100-2000	1100- 18,000	1000- 7000		
Comments		Tests contain level swell and thermal radiation to steam data also	Tests contain level swell data also		

References

1. Anklam, et al., "Experimental Investigations of Uncovered-Bundle Heat Transfer and Two-Phase Mixture Level Swell Under High Pressure Low Heat Flux Conditions," NUREG/CR-2456, ORNL, March 1982.

2. H. Yeh, et. al., "Heat Transfer Above the Two-Phase Mixture Level Under Core Uncovery Conditions in a 336 Rod Bundle," EPRI NP-2161, December 1981.

TABLE H-9
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT TRANSFER–RADIATION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Heat transfer–radiation**	Affects the peak clad temperature (in BWR/2)			
	Plant Range	Facility range			
		Separate effect tests		Integral tests	
Key Parameter/Facility		GOETA (Ref. 1)	THTF (Ref. 2)	TLTA-5A (Ref. 3)	
$T_w - T_v$ (K)	~400	550 –850	<400	400 –600	
Emissivity (-)	0.6 – 1.0	0.7	0.4 – 0.6	~0.6	
Geometry	rod-to-rod and wall	rod-to-rod and wall	rod-to-rod and wall	rod-to-rod and wall	
Comments		stagnant steam	steady-state boiloff	LBLOCA/no ECC	

References

1. Test 27: Experimental investigations of cooling by top spray and bottom flooding for a BWR, Studsvik/RL-78/59, June 1978.
2. Test 3.09.10K: Experimental investigations of uncovered bundle heat transfer…, NUREG/CR-2456.
3. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.

T_w=wall temperature
T_v=steam temperature

TABLE H-10
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT–STORED

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood				
PIRT Parameter	**Heat–stored (fuel and metal structures)**				
	Plant Range	Test Facility (See Tables F-10)			
Plant Parameter					
Temp (K)	570 - 1000				
Comments	Metal to volume ratio is an important parameter				

TABLE H-11
PROPOSED BWR EXPERIMENTAL FACILITIES: INTERFACIAL SHEAR

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown/refill/reflood				
Phenomenon/Justification	**Interfacial shear**	Affects two-phase separation (level), entrainment, and pressure drop			
	Plant Range	Facility range			
		Separate effect tests			
Key Parameter/Facility		CISE (Ref. 1)	GE Level Swell (Ref. 2)	TLTA-5A (Ref. 3)	Pericles (Ref. 4)
P (MPa)	0.1 – 7.2	5.0	0.1 – 7.0		0.2 – 0 .4
G_l (kg/m^2-s)		80 – 380		~0 – 360	
G_v (kg/m^2-s)		4 – 310		2.4 – 360	
Void (-)	~0 – 1.0	0.2 – 0.9	0 – 1.0	0.1 – 1.0	0.2 – 0.9
Geometry	bundle, plenum, pipe	round tube	vessel 1ft & 4ft OD	full-scale bundle	bundle
Comments		steady-state flow	flashing/blowdown Test 1004-3 Test 5801-13	steady-state boiloff	steady-state boiloff

References

1. Density measurements of steam/water mixture flowing in tube, CISE-R-291, December 1969.
2. J. A. Findlay and G. L. Sozzi, "BWR Refill-Reflood Program - Model Qualification Task Plan," General Electric Company document NUREG/CR-1899 (October 1981).
3. Test 6441: BWR BD/ECC program, NUREG/CR-2229.
4. Study of two-dimensional effects in core of LWR during the reflood phase, CEC, Final Report Contract No. SR) 2F, 1984.

Note: Refs. 3 and 4 are applicable for assessment of interphase drag in bundles.

G_l=mass flux of liquid phase
G_v=mass flux of steam

TABLE H-12
PROPOSED BWR EXPERIMENTAL FACILITIES: REWET

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Rewet**	Determines the transition from film to nucleate boiling			
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		TLTA-5A (Ref. 1)			
T_{wall} (K)	650 – 850	620 – 850			
T_{sat} (K)	~550	~550			
P (MPa)	7.0 – 6.0	~6.5			
x (-)	~0.5	~0.5			
Geometry	channeled 8x8 bundle	full-scale bundle			
Comments		LBLOCA/no ECC			

References

1. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.

TABLE H-12 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: REWET

Plant	BWR					
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)					
Transient Phase	Reflood					
Phenomenon/Justification	**Rewet**	Determines the transition from film to nucleate boiling				
	Plant Range	Facility range				
		Separate effect tests				
Key Parameter/Facility		GOETA (Ref. 2)	NEPTUN (Ref. 3)	BWR-FLECHT (Ref. 4)	PWR-FLECHT (Ref. 5)
P (MPa)	0.1 – 1.0	0.7	0.1 – 0.4	0.15 – 0.45	0.15 – 0.30
T_{wall} (K)	600 – 800	850 – 1100	1030 – 1140	1030 – 1220	530 – 1140
$T_{sat} - T_{ECC}$ (K)	25 – 150	75	22 – 134	0 – 90	10 – 80
V_{flood} (cm/s)	2.5 – 10.0	–	1.5 – 15.0	8 – 14.0	1.6 – 3.8
W_{spray} (kg/s) *)	0.5 – 0.75	0.44	–	–	–
Geometry	channeled 8x8 bundle	channeled 8x8 bundle	half-length bundle	7x7 bundle	10x10 bundle
Comments	*) on channel basis	top reflood	bottom reflood	bottom reflood	bottom reflood

References

2. Test 42: Experimental investigations of cooling by top spray and bottom flooding for a BWR, Studsvik/RL-78/59 (June 1978).
3. NEPTUN bundle reflooding experiments, EIR Report No. 386 (1981).
4. Effect of geometry and other parameters on bottom flooding heat transf. associated with nucl. fuel bundle simulators, ANCR-1049 (April 1972).
5. FLECHT – low flooding rate cosine test series, WCAP-8651 (December 1975).

TABLE H-12 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: REWET

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Reflood				
Phenomenon/Justification	**Rewet**	Determines the transition from film to nucleate boiling			
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		TLTA-5A (Ref. 6)	FIST (Ref. 7)		
P (MPa)	0.1 – 1.0	0.1 – 1.0	0.3 – 0.5		
T_{wall} (K)	600 – 800	500 – 800	550 – 800		
$T_{sat} - T_{ECC}$ (K)	25 – 150	132	84 – 104		
V_{flod} (cm/s)	2.5 – 10.0	5.1			
W_{spray} (kg/s) *)	0.5 – 0.75	0.67	0.5		
Geometry	channeled 8x8 bundle	full scale bundle	full scale bundle		
Comments	*) on channel basis	LBLOCA	LBLOCA		

References

6. Test 6424/Run 1: BWR BD/ECC program, NUREG/CR-2229.
7. Test 4DBA1: BWR FIST Phase 2, NUREG/CR-4128 (March 1986).

TABLE H-13
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–CHANNEL BYPASS LEAKAGE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Flow–Channel Bypass Leakage**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Tests 901, 902, 924, 926, 905 (Ref. 1)	FIST Test 6DBA1B (Ref. 2)		
Pressure (MPa)	0.1 – 7.0	0.1 – 7.0	0.1 – 7.0		
Leakage Flow (kg/s)	0 – 1.5		0 – 1.2		
Geometry	Channel bundle	Simulated leakage paths with drilled holes	Prototypical		
Comments		4 channels	one channel		

References

1. Tasaka, et al., ROSA-III Double-Ended Break Test series for a Loss-of-Coolant Accident in a BWR," Nuclear Technology, Vol. 68, pp. 77-93 (January 1985).
2. H. Kumamaru, et. al., "Similarity Study of ROSA-III and FIST Large Break Counterpart Tests to BWR Large Break LOCA," Nuclear Engineering and Design, 103, pp. 223-238 (June 1986).

TABLE H-14
PROPOSED BWR EXPERIMENTAL FACILITIES: COUNTERCURRENT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill/reflood				
PIRT Parameter	**Flow–countercurrent: upper tie plate**				
	Plant Range	Test Facility			
Plant Parameter		Tobin BD/ECC (Ref. 2)	Jones BD/ECC(GE 8x8 bundle data) (Refs. 1, 3)	Naitoh et. al. (Ref. 4)	GOTA BWR ECC Tests (Ref. 5)
P (MPa)	0.1 – 5.0	Near atmospheric	Near atmospheric	Near atmospheric	0.1- 2.0
Steam Flow (gm/s)	90 – 200	36 – 99	0 – 126	43 –83	
Liquid Flow (cm^3/s)	0 – 1000	549 –972	315 – 916	117 – 1033	0.045 – 2.20 Kg/s
Kf^{-}	0 – 2.1		0.0. – 0.8	0.0 – 0.7	
Kg^{-}	0 - 2.1		1.0 – 2.1	1.0 – 2.1	
Water Temp (°C)	40 – 80	Saturated	38 – 96	27 – 97	37 – 97
Comments	Note that the range for Kf and Kg include the range where CCFL exists. Data on a channel basis	Sat. steam/water	Sat. steam	Steam inlet from bundle bottom	Top spray, 64 rods(CCF in bundle pacers, not in tie plate)

References

1. D. D. Jones, “Test Report TLTA Components CCFL Tests,” GE Nuclear Systems Products Division, BD/ECC Program, NEDG-NUREG-23732, (1977).
2. R. Tobin, CCFL Test Results, Phase 1 – TLTA 7x7 Bundle,” GE Nuclear System Products Division, BD/ECC Program, GEAP-21304-5 (1977).
3. D. D. Jones, “Subcooled Countercurrent Flow Limiting characteristics of the Upper Region of a PWR Fuel Bundle,” GE Nuclear Systems Products Division, BD/ECC Program, NEDG-NUREG-23549, (1977).
4. M. Natitoh, et. al., “Restrictive Effect of Ascending Steam on Falling Water during Top spray Emergency core Cooling,” J. of Nuclear Science and Technology, Vol. 15, 11, pp. 806, (1978).
5. “Separate Effects Test Matrix for Thermal Hydraulic Code Validation – Volume I,” Organization for Economic Co-operation and Development Nuclear Energy Agency document OECD/GD(94)82 (September 1993).

TABLE H-14 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: COUNTERCURRENT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill/reflood				
PIRT Parameter	**Flow–countercurrent: upper tie plate**				
	Plant Range	Test Facility			
Plant Parameter		UPTF (Ref. 6)			
P (MPa)	0.1 – 5.0	0.3 – 1.5			
Steam flow (Kg/s)	61 – 153	35 – 300			
Liquid flow (Kg/s)	300 – 460	30 – 1200			
Kf^{-}	0 – 2.1				
Kg^{-}	0 – 2.1				
Water Temp. (°C)	40 – 80	Sat – 30.0			
Flow cross section, (m^2)	2.6	3.755 (1:1 scale)			
Comments	Flow cross section is for BWR/4, hole dia. is also important. Data on a core basis.	Steady-state, hot-leg water injection			

Nomenclature

P, pressure
Kf^{-}, Kutateladze No. for liquid
Kg^{-}, Kutateladze No. for steam

References

6. U. Simon, et al, "UPTF Calibration Tests, Final Report on Research Project BMFT 1500664, Kraftwerk Union, Technischer Bericht
R 54/85/14, December 1985.

TABLE H-15
PROPOSED BWR EXPERIMENTAL FACILITIES: COUNTERCURRENT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill/reflood				
PIRT Parameter	**Flow–countercurrent: side entry orifice**				
	Plant Range	Test Facility			
Plant Parameter		Jones BD/ECC (GE 8x8 bundle data)			
P (MPa)	0.1 – 2.0	Near atmospheric			
Steam flow (gm/s)	7 – 30	0 – 38			
Liquid flow (cm3/s)	0 – 500	0 – 505			
$Kf^{1/2}$	0 – 3.0	0.0 – 1.2			
$Kg^{1/2}$	0 – 1.8	0.9 – 2.0			
Water Temp. (°C)	40 – 80	Saturated steam/water			
Comments	Flow rates are for channel	Bundle bottom inlet, side entry orifices; five orifices sizes			

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. D. D. Jones, “Test Report TLTA Components CCFL Tests,” GE Nuclear Systems Products Division, BD/ECC Program, NEDG-NUREG-23732, (1977).
2. K. H. Sun and R. T. Ferdandez, ”Countercurrent Flow Limitation Correlation for BWR Bundles during a LOCA,” ANS Transactions, Vol. 27, pp. 605 (1977).
3. K. H. Sun, “Flooding Correlations for BWR Bundle Upper tie Plate and side Entry Orifices,” Second Multi-Phase Flow and Heat Transfer Symposium Workshop, Miami Beach, Florida, April 16-19, 1979.

TABLE H-16
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	**Flow–Distribution: lower plenum**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Tests 901, 902, 924, 926, 905 (Ref. 1)	FIST Test 6DBA1B (Ref. 2)	TLTA Tests 6422 Run 3, 6424 Run 1, 6423 Run 3, & 6426 Run 1 (Ref. 3)	SSTF Test EA2-2 (Ref. 3)
Pressure (MPa)	0.1 – 5.0	0.1 – 7.0	0.1 – 7.0	7.1	0.507
					Low plen inj rate = 3.024 kg/s Core steam inj rate= 4.98 kg/s LPCI = 49.21 l/s Subcooling of inj water= 105 K

References

1. Tasaka, et al., ROSA-III Double-Ended Break Test Series for a Loss-of-Coolant Accident in a BWR," Nuclear Technology, Vol. 68, pp. 77-93 (January 1985).
2. H. Kumamaru, et al., "Similarity Study of ROSA-III and FIST Large Break Counterpart Tests to BWR Large Break LOCA," Nuclear Engineering And Design, 103, pp. 223-238 (June 1986).
3. NUREG/CR-2571, "BWR Refill-Reflood Program Task 4.8 – TRAC-BWR Model Qualification for BWR Safety Analysis Final Report," October 1983.

TABLE H-17
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–FORWARD

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Jet pump: forward flow**			Affects coastdown of the core flow	
	Plant Range	Facility range			
		Integral tests		Separate Effects Tests	
Key Parameter/Facility		TLTA-5A (Ref. 1)	FIST (Ref. 2)	LSTF 1/6 Scale Jet Pump (Ref. 3)	Full Scale Jet Pump Data (Ref. 4)
N – Ratio (-)	0.15 – 0.22			2 to -2	0.125 – 0.325
M – Ratio (-)	1.5 – 2.5	2 – 2.25		2 to –2	0.35 – 2.25
Forward flow loss (-)		~4.0	~8.0		
P (MPa)				0.4 –8.16	7.05
Fluid Temp (K)				302 – 562	
Suction Flow (Kg/s)				0 – 13.0	Discharge flow = 300 L/s
Drive Flow (Kg/s)				0 – 4.0	200 L/s
Comments		LBLOCA	LBLOCA		

1. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.
2. Test 6DBA1B: BWR FIST: Phase 1 results, NURG/CR-3711, March 1985.
3. G. E. Wilson, “INEL One-Sixth Scale Jet Pump Data Analysis,” EG& G Idaho, Inc. document EGG-CAAD-5357 (February 1981).
4. A. A. Kurdirka and D. M. Gluntz, “Development of Jet Pumps for Boiling Water Reactor Recirculation System,” Journal of Engineering Power, pp. 7 –12, January 1974.

TABLE H-18
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–MULTIDIMENSIONAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Flow–Multidimensional: upper plenum**	Affects CCFL in the upper plenum and top reflood			
	Plant Range	Facility range			
		Component tests			
Key Parameter/Facility		SSTF/UP (Ref. 1)			
P (MPa)	0.1– 1.0	0.2 – 1.0			
W_{spray} (kg/s) *)	~0.5	0.4 – 0.54			
$T_{sat} - T_{ECC}$ (K)	25 – 150	54 – 145			
W_{steam} (kg/s) *)	0.05 – 0.2	0.09 – 0.16			
Geometry	upper plenum	full scale upper plenum			
Comments	*) on channel basis	spray into 2-phase mix			

References

1. BWR refill-reflood program Task 4.4, NUREG/CR-2786, May 1983.

TABLE H-19
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–REVERSE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Flow–reverse: jet pump**				Affects break flow
	Plant Range	Facility range			
		Component tests		Integral tests	
Key Parameter/Facility		INEL 1/6 jet pump (Ref. 1)		TLTA-5A (Ref. 2)	FIST (Ref. 3)
Reverse flow loss (-)	~0.9			~1.2	~1.3
Comments		Covers wide range of BWR jet pump conditions		LBLOCA	LBLOCA

References

1. G. E. Wilson, "INEL One Sixth Scale Jet Pump Data Analysis," EG&G Idaho, Inc. document EGG-CAAD-5357 (February 1981).
2. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.
3. Test 6DBA1B: BWR FIST: Phase 1 results, NURG/CR-3711, March 1985.

TABLE H-20
PROPOSED BWR EXPERIMENTAL FACILITIES: POWER–3D DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Power–3D distribution**	Affects the peak clad temp. location and channel grouping			
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		ROSA-III (Ref. 1)			
P_{rad} (-)	0.5 – 1.2	1 – 1.4			
Geometry	channeled bundles	4 half-length bundles			
Comments		LBLOCA			

References

1. Test 926: ROSA-III experimental program, JAERI-1307, November 1987.

TABLE H-21
PROPOSED BWR EXPERIMENTAL FACILITIES: POWER–DECAY HEAT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood, long-term cooling				
PIRT Parameter	**Power: Decay Heat**				
	Plant Range	Test Facility (See Table F-20)			
Plant Parameter					
Time (sec)	$0 - 10^{10}$				
Comments					

TABLE H-22
PROPOSED BWR EXPERIMENTAL FACILITIES: PRESSURE DROP

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Pressure drop**	Affects the flow distribution between the shroud and downcomer			
	Plant Range	Facility range			
		Separate effect tests			Integral tests
Key Parameter/Facility		Sher and Greer (Ref. 1)	Muscettola (Ref. 2)	EPRI (Ref. 3)	ROSA-III (Ref. 4)
P (MPa)	0.7 – 7.2	7.6 and 14	6.9	< 0.2	0.7 – 7.2
G (kg/m^2-s)	~30 – 2020	950 – 6780	1145 – 4370	1500 – 2100	~10 – 1100
x (-)		0 – 0.4	0.01– 0.7		
Geometry	bundle, plenum, pipe	rectangular tube	round tube	square tube	4 half-length bundles
Comments		steam-water	steam-water	air-water	LBLOCA

References

1. Boiling pressure drop in thin rectangular chennels, Chem. Symp. Series 23, 61-73, 1959.
2. Two-phase pressure drop – comparison with measurements, AEEW-R-284, 1963.
3. Experimental study of the diversion cross-flow, EPRI NP-3459, Vol. 1, April 1984.
4. Test 926: ROSA-III Experimental Program, JAERI-1307, November 1987.

TABLE H-23
PROPOSED BWR EXPERIMENTAL FACILITIES: PUMP–PERFORMANCE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Pump–performance: recirculation pump coastdown** Determines the core flow				
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		ROSA-III (Ref. 1)	FIST (Ref. 2)		
Torque/Inertia (s^{-2})	38 – 58	~100			
Time (s)	5 – 8		5 – 8		
Geometry	centrifugal pump	centrifugal pump	centrifugal pump		
Comments		LBLOCA	LBLOCA		

References

1. Test 926: ROSA-III Experimental Program, JAERI-1307, November 1987.
2. Test 4DBA1: BWR FIST Phase 2, NUREG/CR-4128, March 1986.

TABLE H-24
PROPOSED BWR EXPERIMENTAL FACILITIES: SPRAY DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Spray distribution**	Affects CCFL and its breakdown at the upper tie-plate			
	Plant Range	Facility range			
		Component tests			
Key Parameter/Facility		SSTF (Ref. 1)			
Sparger height (m)	0.15 – 0.7	0.15 – 0.4			
2-phase level (m)	0 – 1.0	0 –0.4			
Geometry	upper plenum	full scale upper plenum			
Comments		different BWR sprays			

References

1. BWR refill-reflood program Task 4.4, NUREG/CR-2133, May 1982.

TABLE H-25
PROPOSED BWR EXPERIMENTAL FACILITIES: VOID DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown/reflood				
Phenomenon/Justification	**Void distribution/2-phase level**		Determines heat transfer in the core below and above the 2-phase level, timing of jet pump, inlet orifice, and recirc. suction uncovery		
	Plant Range	Facility range			
		Separate effect tests			
Key Parameter/Facility		Frigg (Ref. 1)	TLTA-5A (Ref. 2)	GE Level Swell (Ref. 3)	SSTF/LP (Ref. 4)
P (MPa)	0.1 – 7 .0	~5.0		0.1 – 7.0	0.2 – 1.0
G (kg/m^2-s)		690 – 1500	2.4 – 360		
Void (-)	0 – 1.0	0 – 0.8	0.1 – 1.0	0 – 1.0	0 – 1.0
Geometry	bundle, plenum, annulus	37-rod bundle	full-scale bundle	vessel 1-ft & 4-ft OD	full-scale lower plenum
Comments		steady-state boiling	steady-state boiloff	flashing/blowdown Test 1004-3 Test 5801-13	flashing experiment

References

1. Frigg-2, Hydrodynamic and heat transfer measurements on a full scale 36-rod Marviken fuel element, ASEA and ABB, 1968.
2. Test 6441: BWR BD/ECC program, NUREG/CR-2229.
3. J. A. Findlay and G. L. Sozzi, "BWR Refill-Reflood Program - Model Qualification Task Plan," General Electric Company document NUREG/CR-1899 (October 1981).
4. BWR refill-reflood program Task 4.4, NUREG/CR-2786, May 1983.

TABLE H-25 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: VOID DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown/reflood				
Phenomenon/Justification	**Void distribution/2-phase level**		Determines heat transfer in the core below and above the 2-phase level, timing of jet pump, inlet orifice, and recirc. suction uncovery		
	Plant Range	Facility range			
		Separate effect tests		Integral tests	
Key Parameter/Facility		ANL (Ref. 5)		TLTA-A (Ref. 6)	FIST (Ref. 7)
P (MPa)	0.1 – 7 .0	1.03 – 4.13		0.1 – 7.0	0.1 – 7.0
G (kg/m^2-s)					
Void (-)	0 – 1.0			0 – 1.0	0 – 1.0
Geometry	bundle, plenum, annulus			full scale bundle	full scale bundle
Comments		Subcooled and saturated void; heat flux is 17-100 kW/liter; subcooling is 2-19K, inlet velocity is 1-6 m/s.		LBLOCA	LBLOCA

5. J. F. Marchaterre, “Natural and Forced-Circulation Boiling Studies,” Argonne National Laboratory document ANL-5735 (May 1960).
6. Test 6424/Run 1: BWR BD/ECC program, NUREG/CR-2229.
7. Test 4DBA1: BWR FIST Phase 2, NUREG/CR-4128, March 1986.

TABLE H-26
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–NATURAL CIRCULATION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood, long term coling				
PIRT Parameter	**Flow–natural circulation**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Test NC-1 NC-5 (Refs. 2-3)	FRIGG Test FT 36a 36b, & 36c (Ref. 1)	FIST 6PNCI-4 (Refs. 4-6)	
Pressure (MPa)	0.1- 7.0	7.35, 2.06	1 –7.0	7.0	
Inlet Subcooling (K)	0- 60	0	3 – 58	0.0	
Exit Qual %	10- 80		3 - 73	0-7	
Mass Flux (kg/m2-s)	0.0-1500	100 - 400	195 – 2160	0-1022	
Heat Flux (MW/m2)	0.0-0.555	Core power: 7-20%	0.21-0.89	0.222	
Downcomer Level(m)	1.6	0.6 – 1.7		1 – 1.6	
Comments		The ROSA Nat Circ tests were conducted by changing pressure, core power, and downcomer liquid level (below the scram level) as test parameters			

References

1. O. Nyland, et. al., “Hydrodynamic and Heat Transfer Measurements on a Full Scale Simulated 36-Rod Marviken Fuel Element with Uniform Heat Flux Distribution, FRIGG Loop Project, FRIGG-2g, 1968.

2. K. Tasaka et. al., “Steam Line Break, Jet Pump Drive Line Break and Natural Circulation Tests in ROSA-III Program for BWR LOCA/ECCS Integral Tests,” Eleventh water Reactor Safety Research information Meeting, Gaithersburg, MD, October 24-28, 1983.

3. K. Tasaka, et al., “ROSA-III Double-Ended Break Test Series for a Loss-of-Coolant Accident in a BWR,” Nuclear Technology, Vol. 68, pp. 77-93 (January 1985).

4. “BWR FIST Phase I Results,” NUREG/CR-3711 (March 1985).

5. “BWR FIST Phase II Results,” NUREG/CR-4128 (March 1986).

6. “TRAC-BD1/MOD1–An Advanced Best Estimate Computer Program for BWR Transient Analysis, Volume 4, Developmental Assessment,” EG&G Idaho, Inc., document NUREG/CR-3633 (April 1984).

APPENDIX I

EXPANDED LISTING OF TRAC-B INPUT DECKS FOR COMMON AND BWR-SPECIFIC SETS, IETS, AND PLANTS

Table I-1 lists the available common and BWR-specific TRAC-B SETs input decks. For each facility input deck, a brief description of the facility, test type, test number, and report reference in addition to the latest code version on which the input deck was exercised are provided. Table I-2 lists the available common and BWR TRAC-B IET input decks in the same format. Table I-3 lists the available BWR TRAC-B plant input decks in the same format. Please note that the TRAC-M input processing can also read TRAC-B format input decks.

Table I-1
TRAC-B INPUT DECKS FOR SEPARATE EFFECT TESTS

Facility	Type of Test	Test ID	References	Decks[a]	Comments
Marviken	Critical Flow	Test 15,24	I-1	Mv7c	10 second blowdown. Can model Tests 15 or 24
CISE	Two-phase flow in an adiabatic vertical pipe	CISE-R-291	I-2	Cistbf1	
THTF	Rod-bundle blowdown heat transfer	306.6B, 308.6C	I-3	Thtf366 Thtf386	
Bennett	Dispersed flow film boiling	Test 5358	I-4	Ben5358	
FRIGG	Natural circulation flow test(36-rod bundle)	Run 301016	I-5	Frgns1,frgnt1	
GE Small Vessel	Level swell	Test 1004-3	I-6	Swl8a	
Edwards Pipe	Critical flow	Blowdown test	I-7	Edpga	
Jet Pump	INEL1/6 scale jet pump		I-8	Jp2cbf1	

[a] TRAC-B Version 014 Input Deck

TABLE I-2
TRAC-B INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
TLTA	LB LOCA	Test 6423	I-9	TRAC-B Version 014, Tlta	
FIST	LB LOCA	Test 6DBA1B	I-10	Fist6dba1b	
FIST	SB LOCA	Test 6SB2C	I-10	Fist6sb2c	
FIST	ATWS type event	Test 6pmc2	I-10	Pmc1bc1	

TABLE I-3
TRAC-B INPUT DECKS FOR NUCLEAR POWER PLANTS

Plant	Transient or Accident	References	Decks	Comments
Browns Ferry	LB LOCA	I-11	BFLBLOCA-TRAC-B	A TRAC-M deck also exists
Browns Ferry	SB LOCA	I-12	BFSBLOCA-TRAC-B	A TRAC-M deck also exists
Browns Ferry	1-pump trip transient	I-13	BF1PUMP-TRAC-B	
Browns Ferry	2-pump trip transient	I-14	BF2PUMP-TRAC-B	
Browns Ferry	Feedwater pump trip transient	I-15	BFFWTRAC-B	
Browns Ferry	Generator load rejection transient	I-16	BFGLRTRAC-B	
Peach Bottom	Feedwater pump trip transient	None	PBFWTRAC-B	A TRAC-M deck also exists
Generic BWR/6	Small break LOCA	None	Bwrstra	
Generic BWR/6	Large break LOCA	None	Bwrltra	
Generic BWR/6	Recirculation pump trip	None	Blackfox	
Dresden	Recirculation pump trip	None	Dresden	
Generic BWR/4	Recirculation pump trip	None	Gbwr4lds-1dkin	
Grand Gulf	Steady-state deck	None	Grandg	
LaSalle	85% power with recirculation pump trip and reactivity transient	None	Lasalle	

REFERENCES

I-1. "Marviken Full-Scale Critical Flow Tests-Interim Report Description of the Test Facility," MXC-101, Marviken Power Station, Sweden.

I-2. G. Agnostini, et al., Density Measurements of Steam Water mixtures Flowing in a Tubular Channel Under Adiabatic and Heater Conditions, CISE-R-291, December 1969.

I-3. D. G. Morris, et al., "A Preliminary Evaluation of Rod Bundle Post CHF Heat Transfer to High Pressure Water in Transient Upflow," Interim Report for THTF Test 3.06.6B, ORNL, PWR-DBHT Separate Effects Program, November 1980.

I-4. W. Bennett, Heat Transfer to Steam Water Mixtures Flowing in Uniformly Heated Tubes in which the Critical Heat Flux Has Been Exceeded, AERE-R-5373, 1967.

I-5. O. Nyland, et al., "Hydrodynamic and Heat Transfer Measurements on Full-Scale simulated 36 Rod Marviken Fuel Element with Uniform Heat Flux Distribution," FRIGG Loop Project, FRIGG-2g, 1968. See also O. Nyland, et al., FRIGG Loop Project, FRIGG, R4-494/RL.

I-6. J A Findlay, "BWR Reflood-Reflood Program Model Qualification Task Plan," NUREG/CR-1899 October 1981.

I-7. R. Edwards and F. P. O'Brien, "Studies of Phenomena Connected with the Depressurization of Water Reactors," Journal of the British Nuclear Energy Society, **9**, 1987, pp. 125-135.

I-8. G. E. Wilson, INEL, One-Sixth Scale Jet Pump Data Analysis, EGG-CAAD-5357, February 1981.

I-9. L S Lee et al., "BWR Large Break Simulation Tests-BWR Blowdown Emergency Core Cooling Program," 1982.

I-10. G. Stevens, "Full Integral Simulation Test (FIST) Facility Description Report," NUREG/CR-2576, December 1982.

I-11. UMCP-TRAC-012, "Large Break LOCA TRAC-B Input Model for Browns Ferry Unit 3," January 1998.

I-12. UMCP-TRAC-014, "TRAC-B Analysis of the 0.14 Ft^2 Recirculation Line Break for Browns Ferry Unit 3," January 1998.

I-13. UMCP-TRAC-02, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 One Pump Trip Transient," September 1997.

I-14. UMCP-TRAC-03, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 Two Pump Trip Transient," September 1997.

I-15. UMCP-TRAC-04, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 Feedwater Pump Trip Transient," September 1997.

I-16. UMCP-TRAC-05, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 Generator Load Rejection Transient," October 1997.